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# U. S. REACTOR CONTAINMENT TECHNOLOGY

## A COMPILATION OF CURRENT PRACTICE IN ANALYSIS, DESIGN, CONSTRUCTION, TEST, AND OPERATION

## VOL. II

Wm. B. Cottrell and A. W. Savolainen
Editors

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and Bechtel Corporation

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## Volume II

# 7. DESCRIPTIONS OF SPECIFIC CONTAINMENT SYSTEMS

H. B. Piper*

## 7.1 INTRODUCTION

The philosophy of containment of reactors has changed through the years of development of the nuclear industry. The purpose of this chapter is to briefly relate the story of this change, describe the design and philosophy of existing and recently constructed containers, describe some new and different approaches to the problem of containment, and present tabulated data on the features of existing containment systems. Information for the preparation of this chapter was gained primarily from Hazards Summary Reports (and amendments) for the various reactors discussed. Where necessary, visits to the installations were made to obtain more detailed information and to become familiar with the installation. Information obtained from sources other than these is referenced.

The early experimental reactors had no containment provisions at all. This lack of containment was not necessarily the result of ignorance on the part of the designer but was more the realization that the potential danger from these reactors was insignificant. These reactors were housed in conventional buildings, which were provided primarily to shelter the plant and its operators. A typical example is the ORNL Graphite Reactor (Fig. 7.1),[1] a graphite-moderated, natural-uranium reactor constructed in 1943. It was the second critical reactor ever built and was in continuous service until it was retired on Nov. 4, 1963 after 20 years of operation.

As the potential power level of reactors increased and, as a result, the fission-product inventory increased, the philosophy changed from that of sheltering the reactor to one of controlling, in some way, the disposition of the radioactive particles and gases that might be released in case of an accident or fuel element failure. One of the first methods employed in an attempt to minimize the public hazard from a reactor accident was to enclose the reactor in a relatively gastight envelope. The Knolls Atomic Power Laboratory (KAPL) facility at West Milton, N.Y., was one of the first (1953) and remains the largest (6,000,000 $ft^3$) gastight shell ever to be built to contain a reactor (Fig. 7.2). This rapidly became the most used type of containment facility, mainly because it offers the most certain capability of confining the reactor accident with least dependence upon engineering devices that must operate in order to mitigate the effects of the accident. This type of container has been used in both high- and low-pressure applications. The dividing line between high and low pressure is one of definition and is generally placed at approximately 5 psi.

The integrity of a gastight containment system is dependent upon the penetrations (air locks, equipment doors, piping, and electrical lines) that must seal and be relatively leak free at the design pressure of the

*Oak Ridge National Laboratory, Oak Ridge, Tenn.

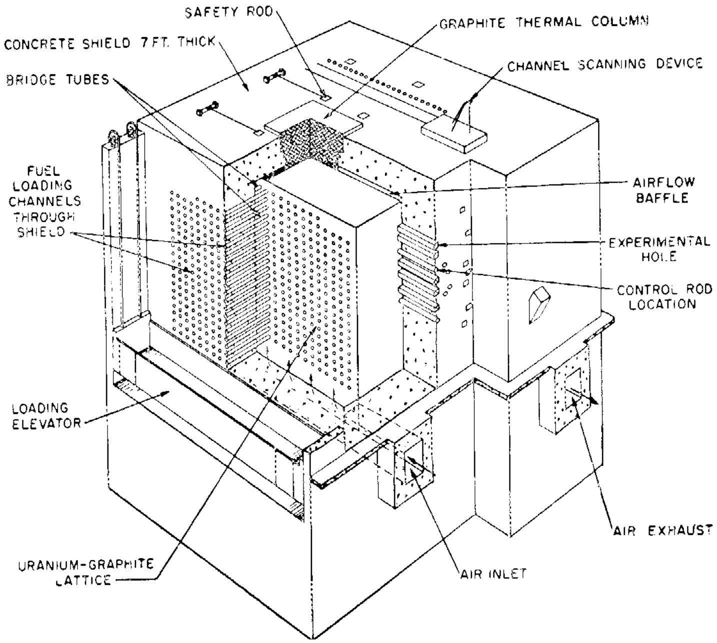



Fig. 7.1. Cutaway Drawing of ORNL Graphite-Moderated Natural Uranium Reactor. (From ref. 1)

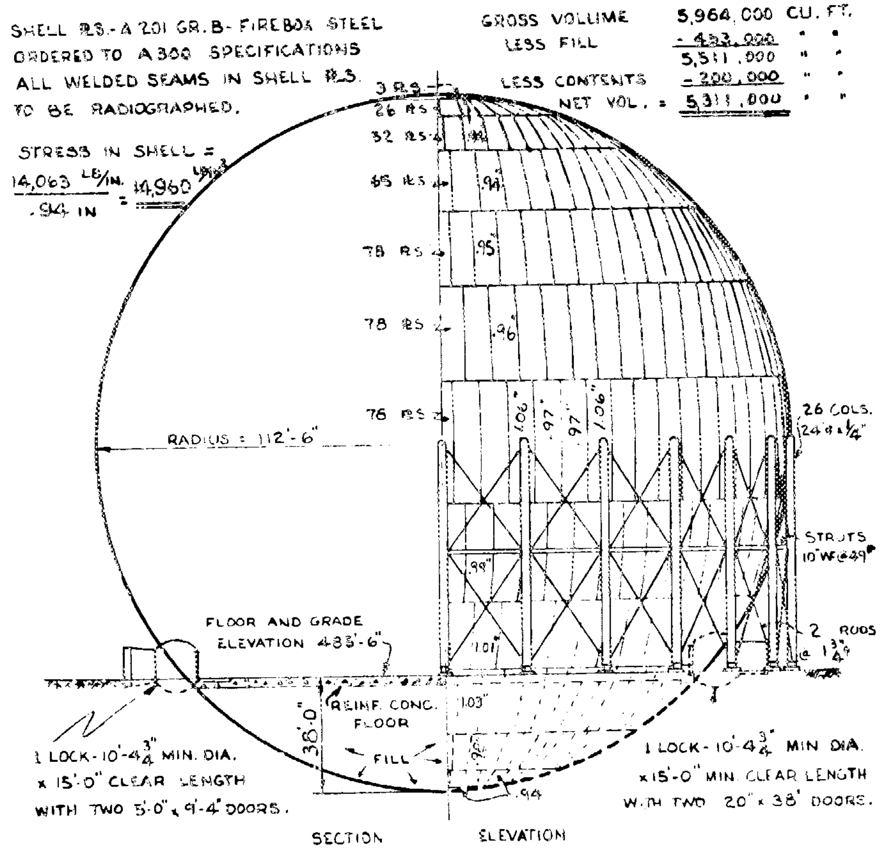



Fig. 7.2. West Milton Containment Vessel. (Courtesy of Chicago Bridge and Iron Company, New York.)

system. In general, as container size or design pressure increases, the problems of this method of containment also increase. In particular, with greater plant complexity, the number of container penetrations increases. As these design requirements increase, the reliability of the system decreases. This fact, together with the desire for improvements in reactor safety, the understanding of reactor safety, and the incentive provided by the goal of economic power, has led to investigation of other, more economical methods of containment.

The Oak Ridge Research Reactor (ORR) was constructed at ORNL about the same time that the first gastight reactor container was built at KAPL. The ORR is housed in a conventional building (Fig. 7.3) that is maintained at a slightly negative pressure. The pressure is maintained by continuously discharging building air through filters to a stack. In the event of accidental release of radioactivity, the gaseous effluent would pass through the filters and the stack. It is considered that this is an economical method of containing some reactor systems, especially those that would not have a large energy release as part of the maximum accident or those for which the energy release and activity release processes would be separated in time.

Another type of containment system[2] provides for pressure suppression (Fig. 7.4). This system is especially suitable for use with boiling- or pressurized-water reactors in which the maximum accident is one involving the release of a great amount of energy in the form of steam from the reactor primary system. This steam is directed from the reactor vessel container, i.e., drywell, through ducts, the discharge ends of which are submerged under a few feet of water. The steam, in bubbling through the water, is condensed, and fission fragments may be scrubbed out. Thus

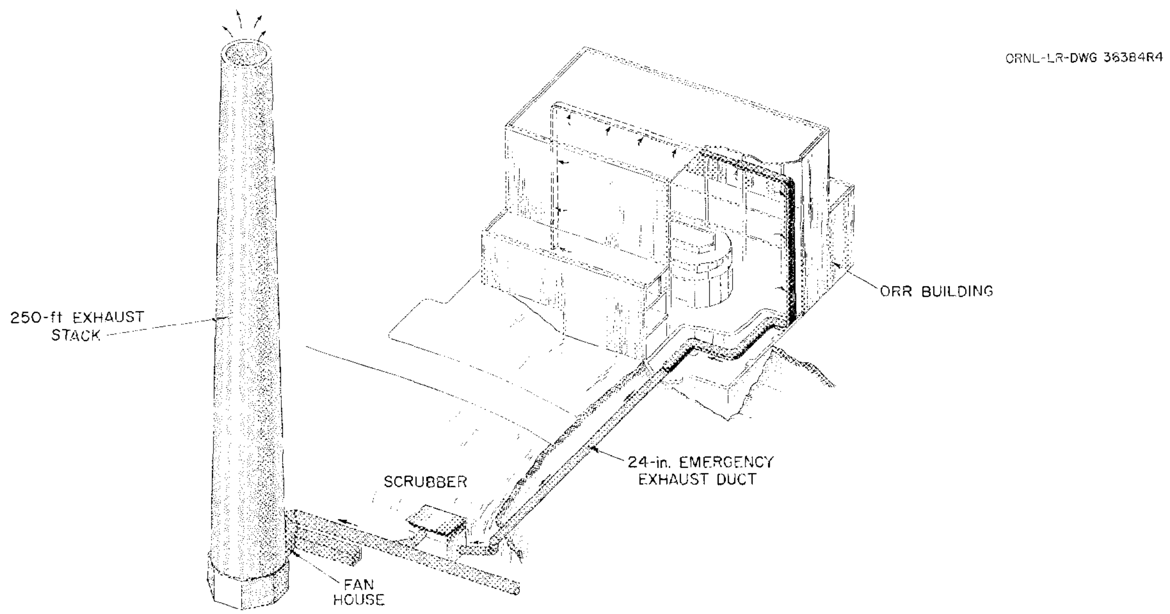



Fig. 7.3. ORR Negative-Pressure Containment System.

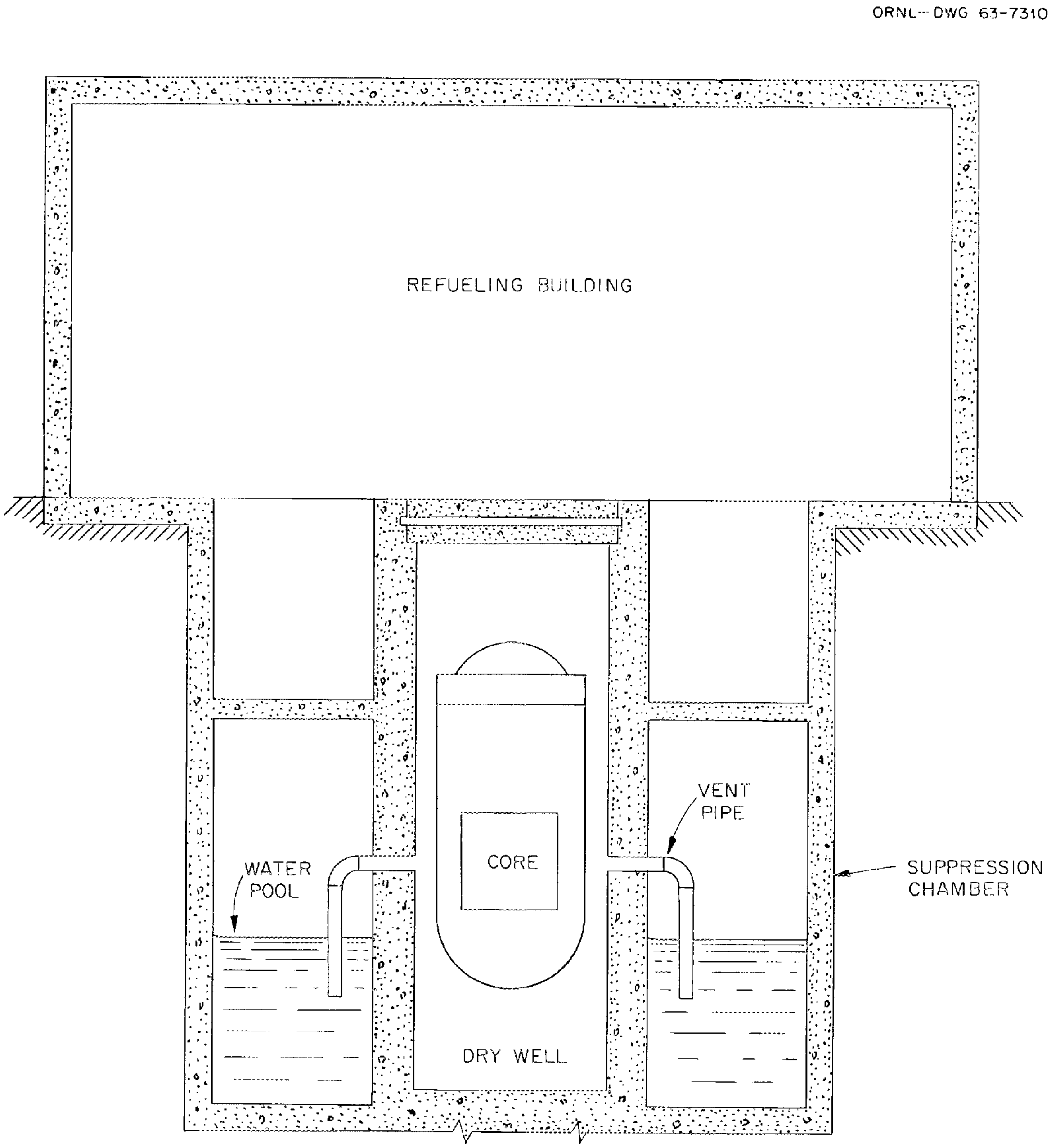



Fig. 7.4. Simplified Pressure-Suppression Containment System. (From ref. 2)

the containment problem is greatly reduced and may be dealt with in a more conventional manner, such as by a low-pressure gastight envelope or by a negative-pressure system.

Pressure relief is another scheme that has received attention recently. The system would allow the initial pressure surge that accompanied the design accident to be vented directly to the atmosphere under the assumption that a negligible quantity of fission fragments would be released initially. After the pressure peak had abated (a matter of seconds or minutes), the vents would be valved closed so that the fission fragments subsequently released when (if) the fuel melted would be confined. Subsequent rises in pressure could be controlled by a spray-dousing system or by controlling the airflow inside the sealed building. For this type of confinement, the building must be constructed to more strict specifications than those for an ordinary mill structure, but the requirements are less stringent than those for the high-pressure container. The New Production Reactor at Hanford uses such a container.[3]

Cavern containment is being used in some European countries where the underlying rock formations are amenable to this application. This idea for containment has been considered but not yet used in the United States. Figure 7.5 shows a typical plan and elevation view of a cavern containment system.[4]

### 7.1.1 General Design Considerations

There are many problems and considerations common to practically all containment systems. Components must meet system specifications as to size and capability; the types of accidents that may occur must be considered; the penetrations and closures must be designed; the tests to be used to measure containment leakage must be determined; and the maximum credible accident (mca) must be defined and described.

The topic of size must be considered early in the design. It is a function not only of the physical dimensions of the building that will house the reactor plant but also of the type of containment to be employed. For instance, with pressure suppression the ductwork is important because of the turbulent pressure drop as a result of the escaping air and steam; with pressure-venting containment (controlled airflow) the blowers that provide the suction for the building are the controlling item; with the pressure-relief system the vent ducts must be properly sized to quickly relieve the pressure buildup in the building.

The electrical connections to a power reactor are an important consideration both from the point of view of continuous delivery of reactor power to the network and the availability of power at the plant for the operation of vital equipment during emergencies. The electrical power distribution system in nuclear plants is not discussed in this chapter but is covered in Section 9.9 of Chapter 9.

#### 7.1.1.1 High- and Low-Pressure Containers

The considerations involved in sizing high- and low-pressure containment shells may be discussed together, because they are similar in concept. The difference is only in the convention of naming those designed to contain pressure above 5 psig as high-pressure containers and those designed for up to 5 psig as low-pressure containers.

There are three important parameters involved in selecting a steel-shell containment vessel. These include the mca energy release (pressure), the physical size (free volume and configurations to be contained), and the type of steel plate to be used. As far as the shell plate is concerned, the important considerations are the design stress, the resulting thickness, and the temperature at which the required Charpy V-notch impact values can be obtained (see Sec. 8.5). The latter temperature establishes the minimum operating temperature. To avoid stress-relieving the vessel

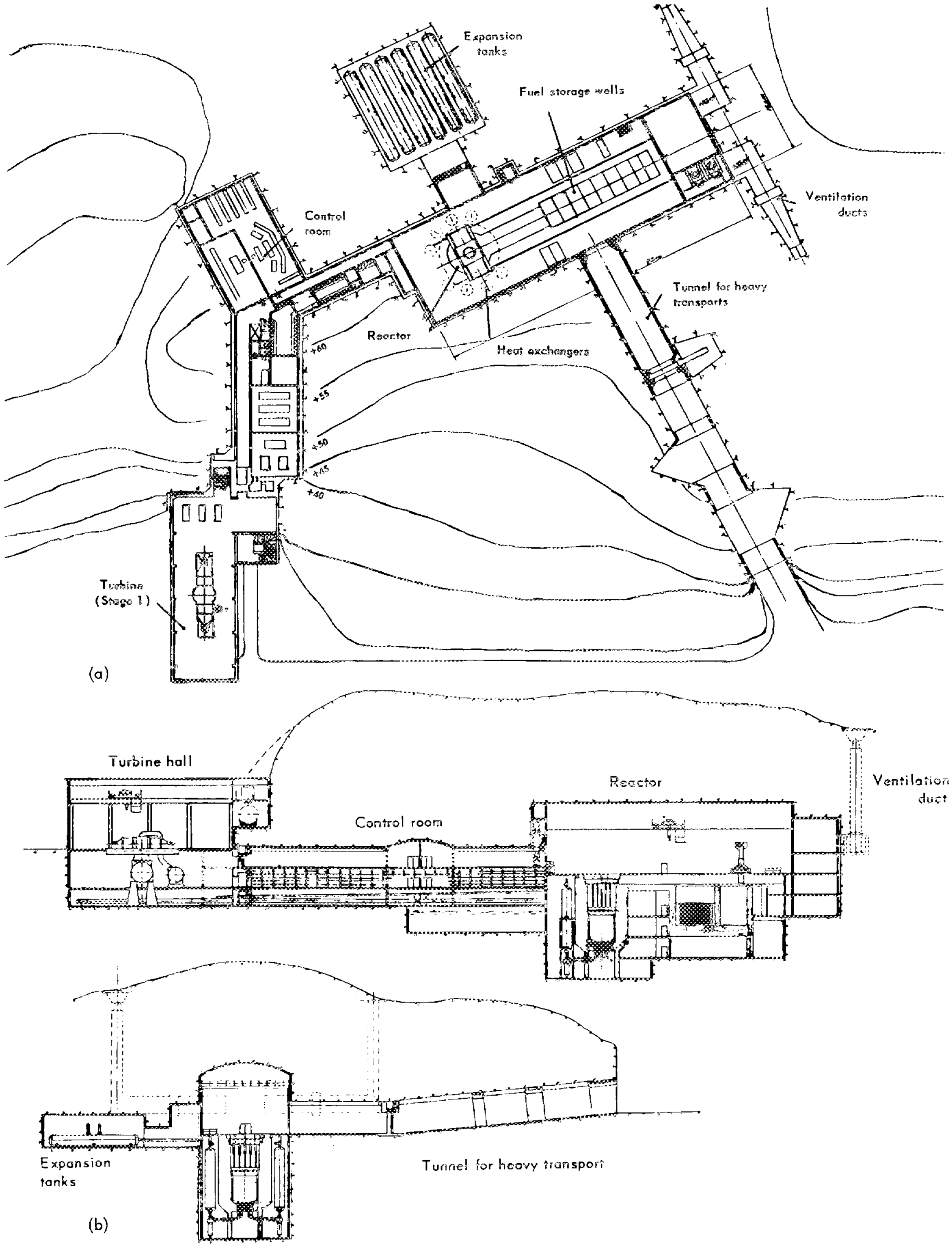



Fig. 7.5. Typical Reactor Container Built in a Cavern Excavated in Rock; the R-3 Plant, Sweden. (a) Horizontal section. (b) Vertical section. (From ref. 4)

upon completion of welding, the thickness of the welded shell joints must not exceed 1.5 in. for the usual steels. The 1.5-in. maximum is established in Table UCS-56 of Section VIII of the ASME Code for steel plates of P-1 classification, and Par. N-1211 of Section III permits such plates when meeting the requirements of SA-300 and the stated impact values. Most containment shells have utilized SA-201 grade B steel plates.

Stress-relieving a completed vessel in the field can increase the cost considerably, although field stress-relieving of large reinforcement assemblies in containment vessels has been successfully accomplished on several occasions by the use of temporary furnaces. To keep the required plate thickness (or butt-welded joint) from exceeding 1.5 in., the cylindrical-shell radius can be limited, or a spherical vessel can be constructed. Theoretically, a sphere requires half the thickness of a cylinder having the same radius and internal pressure. The relation between diameter and pressure for a cylinder and a sphere of SA-201 grade B steel or equivalent with wall thickness of 1.5 in. is shown in Fig. 7.6 (ref. 5). Any vessel whose requirements fall below the curves of Fig. 7.6 may be fabricated without stress-relieving.

Generally speaking, there are fabrication and erection expenses involved with a sphere that are offset to a variable extent by the decreased plate thickness (reduced material and welding costs). There is less wasted space in the cylinder from the equipment placement point of view; therefore, the cylindrically shaped container is a good choice as long as the plate thickness is within the thickness limits and the required Charpy V-notch impact values can be obtained 30 degrees below the service temperature.

The consideration of pressure to be contained and the free volume in which it is contained involves dependent as well as independent factors.

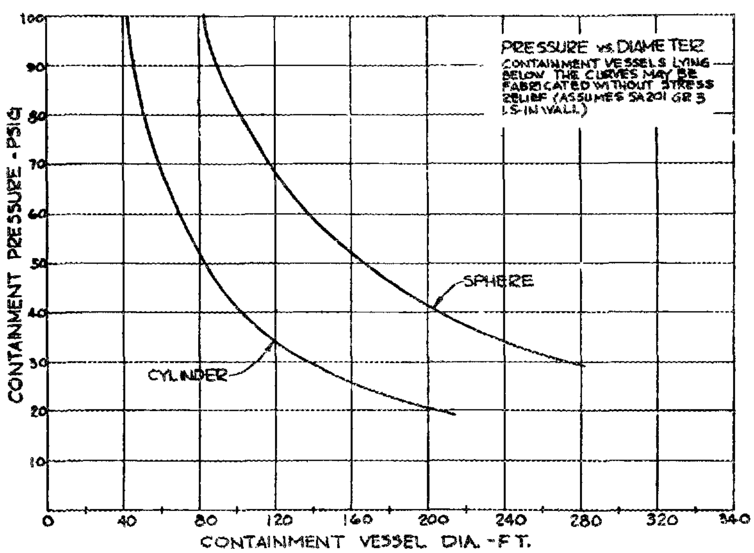



Fig. 7.6. Internal Pressure of Cylinder and Sphere as a Function of Diameter for a Given Energy Release. (From ref. 5)

The amount of energy that might be released as a result of a reactor accident and the minimum free volume required for servicing and maintenance are somewhat independent, but energy and size combine to determine pressure. There are, of course, unlimited combinations of size and pressure that may be used as design parameters for the container. Figure 7.7 is a plot of free volume per pound of released coolant versus final pressure as a function of the average internal energy of the coolant.[5] In this case the coolant is water; and the process of the release is considered to be adiabatic, with perfect mixing of the gases and vapors in the container. If there is any energy addition as a result of a chemical reaction or a nuclear excursion, or loss of energy due to heat transfer or condensation, such effects must be added. Figures 7.6 and 7.7 give, however, a first approximation to container dimensions; after this, refinements may be made that result in a more particular solution to a design problem (see Chap. 6).

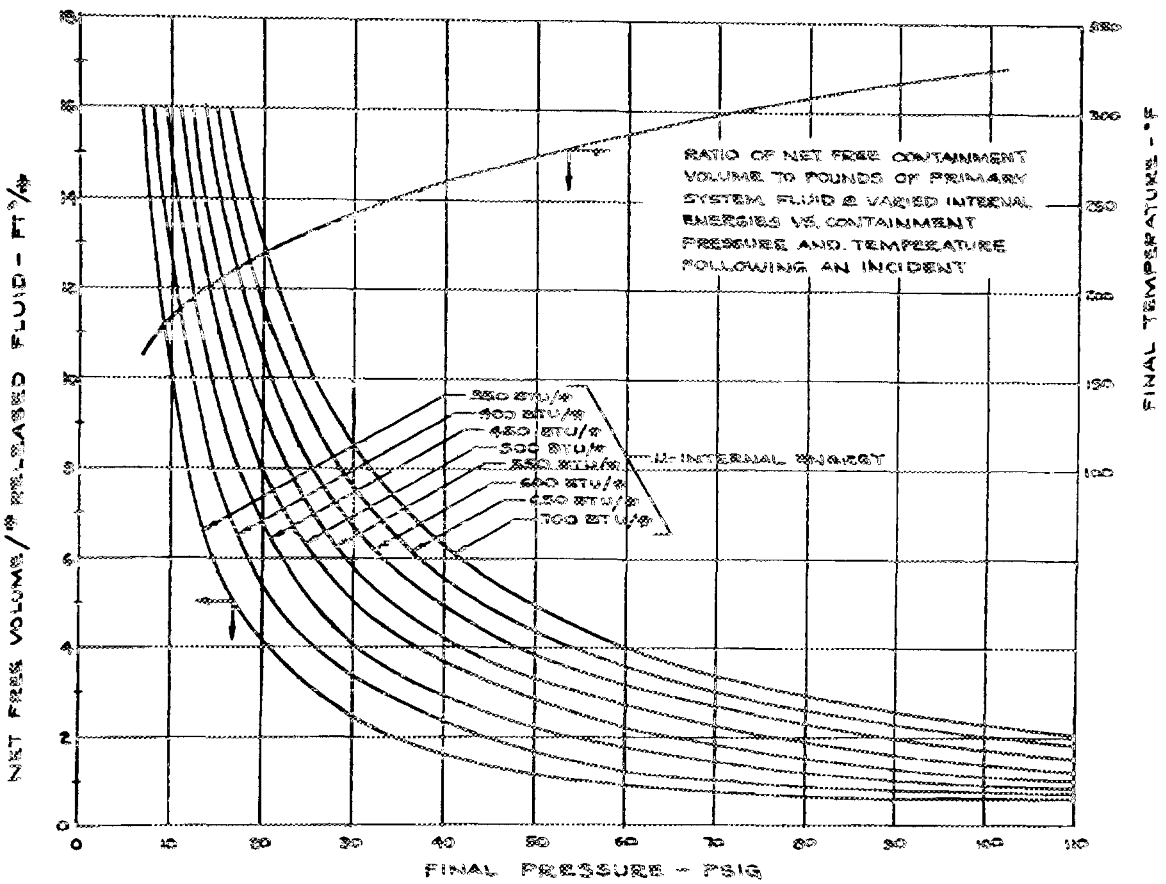



Fig. 7.7 Free Volume and Final Pressure of Containment Shell for a Given Energy Release. (From ref. 5)

### 7.1.1.2 Pressure Suppression

The same general items must be taken into account in the design of a pressure-suppression system as those cited above for a pressure-containment shell. The main difference is that the high peak pressure for which the reactor dry well must be designed will be quickly reduced as the pressure is relieved by large-diameter ducts that discharge the vapor and gas contents of the dry well into the pool of the suppression chamber. Further, the suppression chamber must be designed for minimum leakage under the driving force of the pressure to which it will be subjected.

The dry well of the pressure-suppression system is physically small, since it contains only the reactor pressure vessel and other primary system components (e.g., circulation pumps) connected to the reactor vessel without intervening isolation valves. Small pumps, primary piping (steam generator and pressurizer, if any), and certain valves are outside the dry well but are inside the plant building.

The size of the dry well is dictated by the size of the reactor being contained. The wall thickness of the dry well is determined in the same way as that for the high-pressure container; that is, it is usually assumed that the vessel must take the pressure of the loss-of-coolant accident, with conservative credit being taken for condensation and time dependence of the release. The relief ducts are designed to offer small resistance to flow of the steam-water mixture from the dry well to the suppression pool to limit dry-well pressure. This is a straightforward fluid-flow problem. Experiments have indicated that the pressure in the suppression chamber (above the pool) will be due, primarily, to the dry-well air that has been displaced by the escaping steam (i.e., all the steam is condensed in the suppression pool).

The refueling building houses, or contains, the suppression system and provides shielding in the event of an incident during the refueling operations. If an emergency occurs in the fuel-handling operations (e.g., a dropped fuel element), containment is effected by the pressure-venting system (discussed below), in this case the building is maintained at a negative pressure so that air is leaked in. The building air is filtered and discharged through a high stack.

### 7.1.1.3 Pressure Venting

The containment capability of the pressure-venting system depends on a directed transport of air into the building and the filtration and stack discharge of the building air. When this technique is employed with structures that are maintained below atmospheric pressure, it is frequently called negative-pressure containment. This system has been referred to as the ORR type of containment, mainly because the ORR was the first reactor to use this scheme in the United States. This scheme is not useful for housing a reactor that may be subject to a large and sudden energy release; it is suitable for reactors that have a smaller amount of stored energy, such as some large gas-cooled power reactors and the production reactors at Hanford. These reactors may, under severe conditions, release a large quantity of fission fragments, but there would be little accompanying pressure rise due to the sudden release of stored, nuclear, or chemical energy. The purpose of this containment system is to control, predictably, the disposition of the released fission products. In order to perform this function, the capacities of various components must be carefully designed to be adequate to do the work.

The building must be sized and constructed so that its ventilation system is capable of maintaining in-leakage under all probable atmospheric conditions. The pressure difference must be provided by a system of fans and ductwork that will yield a uniform negative pressure in the building when operating at the specified volumetric flow rate. Next, the filters in the discharged gas stream (caustic, charcoal, etc.) for removing the halogens and particulates must be built to remove, with a specified efficiency, the maximum amount of the constituent that is expected. Finally, a stack of sufficient height and capacity to guarantee safe dispersion of the remaining fission products at the worst conceivable meteorological conditions must be provided.

Several reactor installations use this idea in conjunction with some other basic containment schemes. For instance, the Humboldt Bay unit uses this for its refueling system containment, while its primary or main consideration with regard to accidents is the pressure-suppression system. The Indian Point Reactor, which is contained by a steel sphere, uses this confinement scheme for the air space between the sphere and the concrete building that surrounds it. The NS Savannah employs this principle in the outer portion of its dual containment system.

#### 7.1.1.4 Pressure Relief

A system that would vent the initial pressure surge which accompanied some maximum accident is used for the New Production Reactor at Hanford. For such a system the pressure-relief ports must be designed so that the pressure drop in the port due to the escaping air, gas, and vapor will not impose an intolerable pressure on the building. Similarly, interconnections between vaults or cells within the building must be sized to allow rapid transport of the escaping coolant and steam to the main part of the building.

After a given time the building must be made leaktight; this means that the vent ports, which are designed to allow rapid escape of gases, must be sealed and must resist leakage under the driving force of some given pressure. (These vent ports must be physically large to accomplish this function.) Further, in order to control the long-term pressure in the building, a spray or water-fog system that is capable of maintaining pressure control in the building with excellent dependability must be provided.

### 7.1.2 Example of Containment System Planning

In the design of the pressurized-water reactor plant (the PWR) that was built at Shippingport, Pennsylvania, it was first considered desirable to place the system underground in order to utilize the earth as shielding from direct radiation. However, the existing water tables limited the depth of the lower level, and this factor, together with the expense of excavation, ruled out an underground location. A cylindrical vessel with its long axis parallel to the ground was then selected. It was decided that the maximum pressure should be about 50 psig (as it turned out the design pressure was 52.8 psig) and that the plate thickness should be 1.25 in. (for stress relief). These considerations resulted in a diameter of 50 ft for the container. In this case the pressure, shape, and diameter of the vessel were specified; so the length had to be determined according to the energy to be contained. The pressure limitation would have required a vessel about 300 ft long, which would have provided adequate volume for the contained equipment. It was obvious, however, that fabrication, support, and thermal expansion problems would have been severe in such a vessel. Therefore it was decided to use several such vessels interconnected by large-diameter ducts (see Sec. 7.2, Fig. 7.25). The several containers had to satisfy other requirements with regard to equipment arrangement: (1) the reactor compartment had to be below a canal to facilitate underwater refueling, (2) the boilers had to be higher than the reactor vessel to provide natural convection and a symmetrical arrangement was desirable, and (3) auxiliary equipment needed to be close to the reactor. The arrangement selected consists of four vessels: two are cylinders 50 ft in diameter and 97 ft long, one is a cylinder 50 ft in diameter and 147 ft long, and the other is a 38-ft-diam sphere that contains the reactor.

## 7.2 HIGH-PRESSURE CONTAINMENT

Since many reactors have been housed in high-pressure containers, more experience has accumulated with this type of containment than any other. Available information on 18 selected reactors that have high-pressure containment vessels is summarized in Tables 7.1 through 7.13. These tables are followed by discussions of peculiarities of the various systems. The reactors chosen for review include the following types: pressurized water, boiling water, gas cooled, homogeneous, fast breeder, and pressure tube. The containers are of various shapes.

The fact that the high-pressure containment system is designed to withstand the entire design accident without rupturing makes this system applicable for any type of reactor. Consequently, almost every type of reactor has been or is being contained in a high-pressure envelope, with the notable exceptions being the pool-type reactors for which much less stringent containment capabilities are required. The installations at Indian Point and the NS Savannah are included in this discussion because their inner containment vessels are of the high-pressure type. These plants are, in reality, doubly contained and will be further discussed from that aspect in Section 7.8, "Multiple Containment." Table 7.1 lists the installations discussed here.

High-pressure containers have been built in a variety of shapes and sizes and with several different structural materials and techniques. The size and shape of the container are determined usually by a compromise between several requirements. The container should offer sufficient room for equipment without a large amount of wasted space, and the physical size should be such that the pressure to be contained is not excessive.

Table 7.1. High-Pressure Containment Vessels

| Reactor | Name | Location | Thermal Power (Mw) | Type | Prime Contractor | Architect-Engineer | Containment Fabricator | Nuclear Equipment Supplier | Operator |
|---|---|---|---|---|---|---|---|---|---|
| Big Rock Point | Big Rock Point Plant | Charlevoix, Mich. | 240 | Boiling water, power | Bechtel | Bechtel | CB&I[a] | General Electric | Consumers Power Co. |
| CVTR | Carolinas-Virginia Tube Reactor | Parr, S.C. | 63 | Pressurized tube, power | Westinghouse | Stone & Webster | | Westinghouse | Carolinas-Virginia Nuclear Power Associates, Inc. |
| Dresden | Dresden Nuclear Power Station | Morris, Ill. | 626 | Boiling water, power | General Electric | Bechtel | CB&I | General Electric | Commonwealth Edison |
| Elk River | Elk River Reactor | Elk River, Minn. | 72.7 (58.2 + 14.5) | Boiling water nuclear superheat, power | Allis-Chalmers | Sargent & Lundy | CB&I | Allis-Chalmers | Rural Cooperative Power Associates |
| Enrico Fermi | Enrico Fermi Atomic Power Plant | Lagonna Beach, Mich. | 200 | Fast breeder, power | UE&C | Commonwealth Associates | CB&I | Various | Power Reactor Development Corp. |
| EBR-II | Experimental Breeder Reactor II | NRTS, Idaho | 62.5 | Fast breeder, power | Argonne National Laboratory | H. K. Ferguson | Graver Tank and Manufacturing Co. | Argonne National Laboratory | Argonne National Laboratory |
| EGCR | Experimental Gas-Cooled Reactor | Oak Ridge, Tenn. | 85 | Gas-cooled, power | AEC-ORO | Kaiser | Pittsburgh-Des Moines | Allis-Chalmers | Tennessee Valley Authority and AEC |
| HWCTR | Heavy Water Components Test Reactor | Aiken, S.C. | 61 | Heavy water, test | | | CB&I (steel portion) | | AEC, Du Pont |
| HRT | Homogeneous Reactor Test | Oak Ridge, Tenn. | 5.2 | Aqueous homogeneous, power | ORNL | ORNL | CB&I | Newport News Shipbuilding – pressure vessel and core tank<br>Foster-Wheeler – heat exchangers<br>Westinghouse – pumps | ORNL |
| Indian Point | Consolidated Edison Thorium Reactor | Indian Point, N.Y. | 585 | Pressurized water, power, thorium converter | Owner | Owner | CB&I | Babcock & Wilcox | Consolidated Edison |
| NS Savannah | Nuclear Ship Savannah | (mobile) | 69 | Pressurized water | Babcock & Wilcox | Geo. C. Sharp, Inc. & New York Shipbuilding Corp. | New York Shipbuilding Corp. | Babcock & Wilcox | American Export-Isbrandtsen Lines for the U.S. Government |
| Pathfinder | Pathfinder Atomic Power Plant | Sioux Falls, S.D. | 199.6 (175.2 + 42.4) | Boiling water nuclear superheat, power | Allis-Chalmers | Pioneer Service and Engineering | Pittsburgh-Des Moines | Allis-Chalmers | Northern States Power Co. |
| PRTR | Plutonium Recycle Test Reactor | Hanford, Wash. | 70 | Heavy water, test | | | | | |
| Saxton | Saxton Nuclear Experimental Reactor | Saxton, Pa. | 20 | Light water, experimental | | | Bethlehem Steel | | |
| Shippingport | Shippingport Atomic Power Station | Shippingport, Pa. | 231 | Pressurized water, power | Westinghouse | Stone & Webster | Pittsburgh-Des Moines | Westinghouse | AEC, Duquesne Light Co. |
| SM-1 | Stationary Medium Power | Fort Belvoir, Va. | 10 | Pressurized water, power | Alco | | Bethlehem Steel | | U.S. Army |
| VBWR | Vallecitos Boiling Water Reactor | Pleasanton, Calif. | 30 | Boiling water | Bechtel | Bechtel | Consolidated Steel | General Electric | General Electric |
| Yankee | Yankee Atomic Electric Co. | Rowe, Mass. | 600 | Pressurized water, power | Owner | Stone & Webster | CB&I | Westinghouse | Yankee Atomic Electric Co. |

[a]Chicago Bridge and Iron.

An economic balance must be struck between size, shape, material, and method of construction, with the material and method of construction being dictated in general by the design pressure. Table 7.2 gives the size, shape, design pressure, temperature, and conventional load factors, such as wind, snow, and seismic loading, considered for the plants listed in Table 7.1.

The internal pressure to be withstood usually determines the thickness of material to be used and the physical size of the container. The dependence of free volume on internal pressure is strictly a pressure-volume rate relation whose independent variable is the amount of energy to be contained if the rate at which heat is released to, and removed from, the system is not considered (i.e., the adiabatic case is considered). In general, large water-cooled reactors have a large amount of stored energy and thus require either a large container designed for a reasonable pressure or a reasonably sized vessel designed to contain a high pressure.

Vessels that are required to have a very large free volume are spherical because of economics (see Table 7.2). The sphere requires less material to contain a given pressure than a building of another type of configuration. One notable exception is the EGCR containment vessel, which has a very large contained volume but a relatively low pressure of 9 psig. Most of the other vessels are cylinders with hemispherical tops and either hemispherical, hemiellipsoidal, or flat bottoms. Exceptions are discussed individually below.

The entries in Table 7.2, column 6, regarding design containment temperature may be misleading. The numbers quoted are the final temperatures, but the point of interest here is really the change in temperature during the accident, which would cause expansion of structural members that should be analyzed for excessive stresses. Usually some initial temperature is assumed in the containment vessel, and the temperature difference is obtained by subtraction from the number quoted as the final temperature. The absolute value of the temperature becomes significant (as far as the containment structure is concerned) only if it is above approximately 600°F, which is the temperature at which substantial changes in material properties begin to occur. However, temperatures below 600°F are of interest to those who design any contained equipment that includes thermally sensitive materials, such as electric motors and gaskets.

Since these vessels are constructed to prevent outleakage under the influence of considerable positive internal pressure, the designer does not (and should not) include the strength necessary to resist crushing by the action of a negative internal pressure. Thus negative-pressure protection (column 7, Table 7.2) is of concern. In the event of a negative internal pressure, which might be brought about by condensation of steam after a major accident, some relief device must be available that would be much more reliable than the ordinary "vacuum breaker." The NS Savannah has the highest external pressure specification. This is a special case, since this container could sink into the depths of the sea. If, however, the vessel sank deeper than 100 ft, the bolts on the manway hatches would elongate and fail. This would allow the water to rush in and the pressure to be equalized. After pressure equalization was attained, springs would close the hatch to reseal the vessel.

Table 7.2. Design Parameters of High-Pressure Containment Vessel

| Reactor | Shape | Free Volume ($ft^3$) | Dimensions (ft) | Design Pressure[a] (psig) | Design Temperature (°F) | Negative-Pressure Protection | Conventional Load Factors | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | Wind | Snow | Seismic (× g) | Other |
| | | $\times 10^3$ | | | | | | | | |
| Big Rock | Sphere, partially below grade | 922 | 130 | 27 (-1.22) | 235 | Yes | 30 $lb/ft^2$ | 40 $lb/ft^2$ | 0.05 | |
| CVTR | Reinforced concrete cylinder, partially below grade, with steel dome-top | 243 | 58 d[b]<br>119 h[c] | 21 | 220 | No | 84 mph | | 0.1 | |
| Dresden | Sphere, partially below grade | 2880 | 190 | 29.5 (-1.0) | 325 | Yes | 110 mph | 25 $lb/ft^2$ | 0.033 | |
| Elk River | Vertical cylinder with hemispherical top and hemiellipsoidal bottom, partially below grade | 287 | 74 d<br>115 h | 21 (-0.33) | 220 | Yes | 30 $lb/ft^2$ | 30 $lb/ft^2$ | | |
| Enrico Fermi | Vertical cylinder with hemispherical top and hemiellipsoidal bottom, partially below grade | 280 | 72 d<br>120 h | 32 (-2) | 460 (at building wall) | Yes | 60 $lb/ft^2$ | 30 $lb/ft^2$ | 0.1 | 10 $lb/ft^2$ |
| EBR-II | Vertical cylinder with hemispherical top and hemiellipsoidal bottom, partially below grade | 450 | | 24 | | ? | | | | |
| EGCR | Vertical cylinder with hemispherical ends, partially below grade | 1360 | 114 d<br>216 h | 9 | 200 | Yes | 40 $lb/ft^2$ (top)<br>30 $lb/ft^2$ (cyl.) | 10 $lb/ft^2$ | 0.05 (hor.)<br>0.025 (ver.) | 1 $lb/ft^2$ (insulation) |
| HRT | Rectangular parallelepiped | 24.7 | 54 × 30.5 | 30 (-7.5) | 270 | No | No external loads considered | | | |
| HWCTR | Prestressed concrete cylinder below grade, steel cylinder with dome top above grade | 320 | 70 d<br>125 h | 24 (-0.75) | 226 | Yes | 100 mph | | 0.10 (hor.)<br>0.025 (ver.) | |
| Indian Point | Sphere, partially below grade | 1800 | 160 | 25 (-1.25) | 220 | Yes | Totally enclosed | | | |
| Shielding building | Sphere enclosed in concrete building | 165 | | | | Yes | ASA 58 | ASA 58 | 0.05 | |
| NS Savannah | Horizontal cylinder with spherical heads | 32.3 | 35 d<br>50 h | 186 (-100 ft of sea-water) | 360 | Yes (manway will fail at 100 ft of sea water) | No external loads considered, but loads due to ship motion are considered | | | |
| Pathfinder | Vertical cylinder with hemispherical top and hemiellipsoidal bottom, partially below grade | 145 | 50 d<br>120.5 h | 78 (-3) | 320 | Yes | 30 $lb/ft^2$ | 35 $lb/ft^2$ | UBC[d] | UBC[d] |
| PRTR | Vertical cylinder with hemispherical top and hemiellipsoidal bottom, partially below grade | 400 | 80 d<br>121.5 h | 15 (-0.58) | 205 | Yes | | | | |
| PWR | Three cylindrical containers with hemispherical ends and one spherical container | 473 | Multiple containers[e] | 52.8 (-3.0) | 280 | No | No external loads considered | | | |
| Saxton | Vertical cylinder with hemispherical top and hemiellipsoidal bottom, partially below grade | 141.5 | 50 d<br>109.5 h | 30 | | Yes | 80 mph<br>20 $lb/ft^2$ | 25 $lb/ft^2$ on slope 0–50% | | |
| SM-1 | Vertical cylinder with hemispherical ends, partially below grade | 32.8 | 36 d<br>64 h | 66 | 273 | No | 20 $lb/ft^2$ | | | |
| VBWR | Vertical cylinder with hemispherical ends, partially below grade | 125 | 48 d<br>99 h | 45 (-2) | 292 | ? | 100 mph | | 0.134 | |
| Yankee | Sphere, above grade | 860 | 125 | 34.5 | 249 | No | 100 mph | | | |

[a]The numbers in parentheses represent the design limit for negative internal pressure.

[b]Diameter.

[c]Height.

[d]Uniform Building Code.

[e]Two boiler chambers 50 ft in diameter, 97 ft in length; one auxiliary chamber, 50 ft in diameter, 147 ft in length; one sphere, 38 ft in diameter (see sec. 7.2, Fig. 7.25).

### 7.2.1 Spherical Containers

The Dresden (Fig. 7.8, ref. 6), Indian Point (Fig. 7.9, ref. 7), and Big Rock Point (Fig. 7.10, ref. 8) containment vessels are typical steel spheres that have as their foundations concrete pads which surround the lower one-fourth of the sphere. The outer portions of the steel are in contact with the supporting concrete, except for an intervening coat of an anticorrosion substance, such as paint, Bitumastic, or epoxy. Concrete foundations for the reactor and other heavy equipment are poured inside the shell in direct contact with it.

In general, the container is supported by the bottom foundation and by columns that support the vessel at the horizontal girth line. The inside hardware is supported on foundations that are, in turn, supported by the foundation pad.

#### 7.2.1.1 Yankee

The Yankee containment shell is quite different from the typical steel sphere, both in the manner in which the shell is supported and in the way that the equipment foundations are supported. The bottom of the shell is 24 ft above grade. The vapor container is supported by steel columns attached at the horizontal girth line. The concrete work (foundations for equipment, shielding, etc.) is completely independent of the containment shell. The inner concrete work is supported by reinforced concrete columns that penetrate the sphere through convoluted steel expansion joints. These expansion joints are welded to the shell at one end and are integral with the concrete support columns at the other (see Fig. 7.11, ref. 9).

#### 7.2.1.2 Indian Point

The Indian Point reactor containment structure is peculiar in that the conventional steel sphere is itself contained within a 5 1/2-ft-thick concrete cylinder (Fig. 7.9) with an arched top of concrete blocks 2 3/4 ft thick. The purpose of this is to provide shielding in the event of a maximum accident. Prestressing wires are wound near the top of the cylindrical portion of the shield to give additional strength for support of the arched concrete (prestressed) beams that carry the load of the roof blocks.

The concrete building would be further utilized in the event of an accident to reduce the amount of fission-product activity available to be released to the atmosphere. The annular space is maintained at a slightly negative pressure, and the air is routed through filters and up the stacks. This containment scheme is further discussed in Section 7.8.

#### 7.2.1.3 Dresden

The containment design basis was required for Dresden (Fig. 7.8) prior to the establishment of the maximum credible accident concept. In order to proceed with containment structure procurement, a design pressure

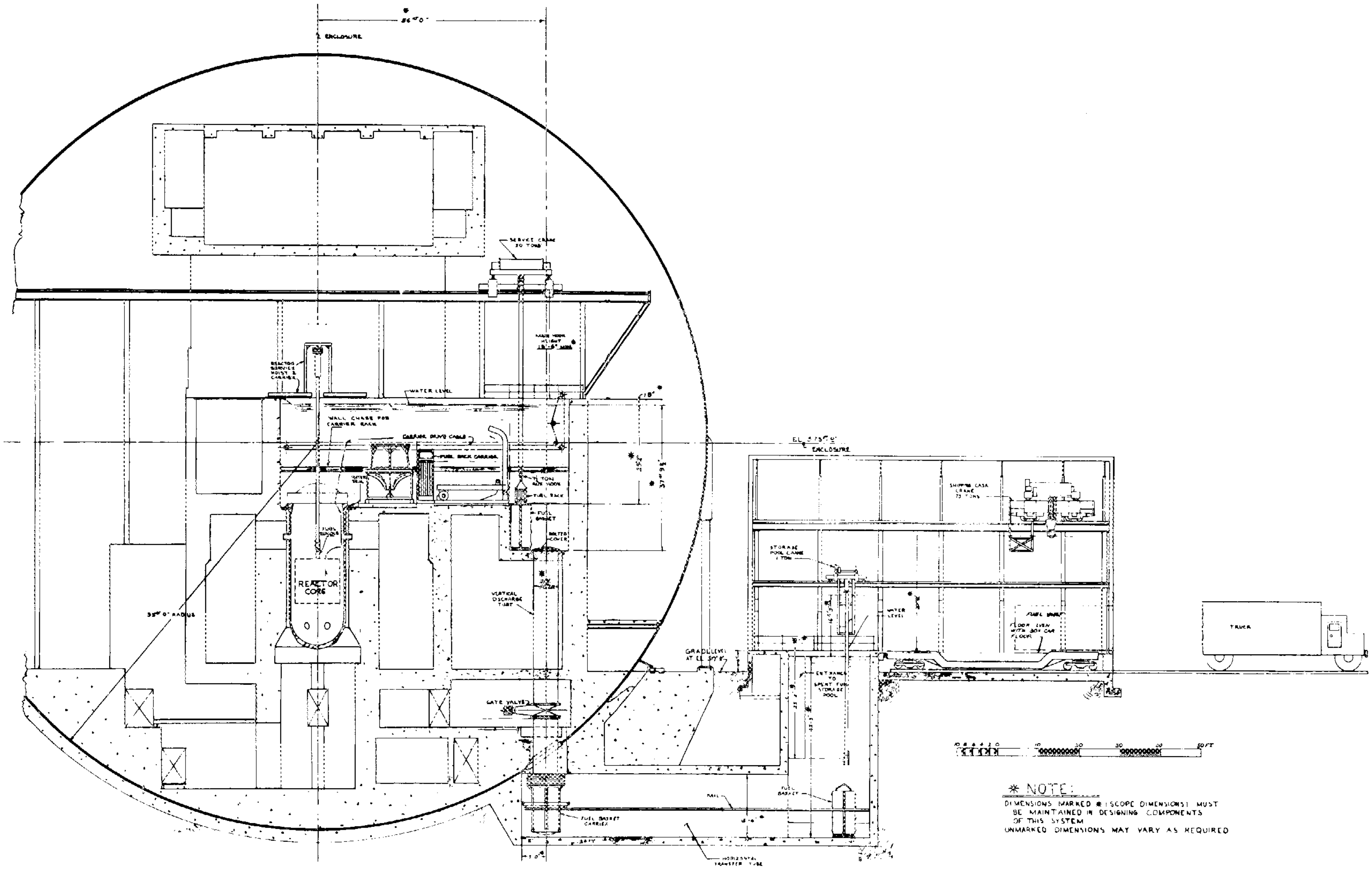
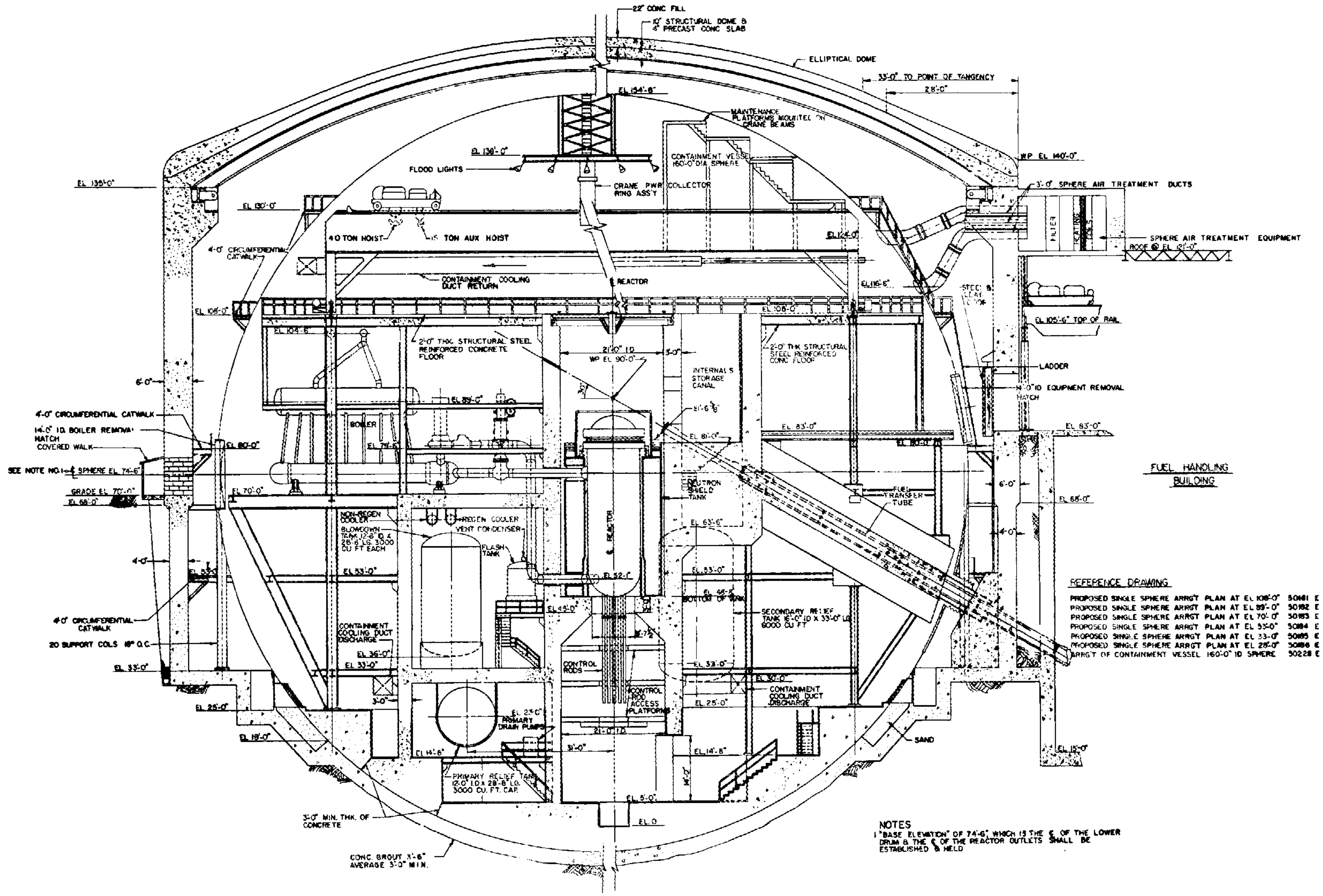



Fig. 7.8. Containment Vessel of the Dresden Nuclear Power Station. (From ref. 6)




SECTION "B-B"

Fig. 7.9. Containment Vessel of the Consolidated Edison Thorium Reactor at Indian Point, N.Y. (From ref. 7)

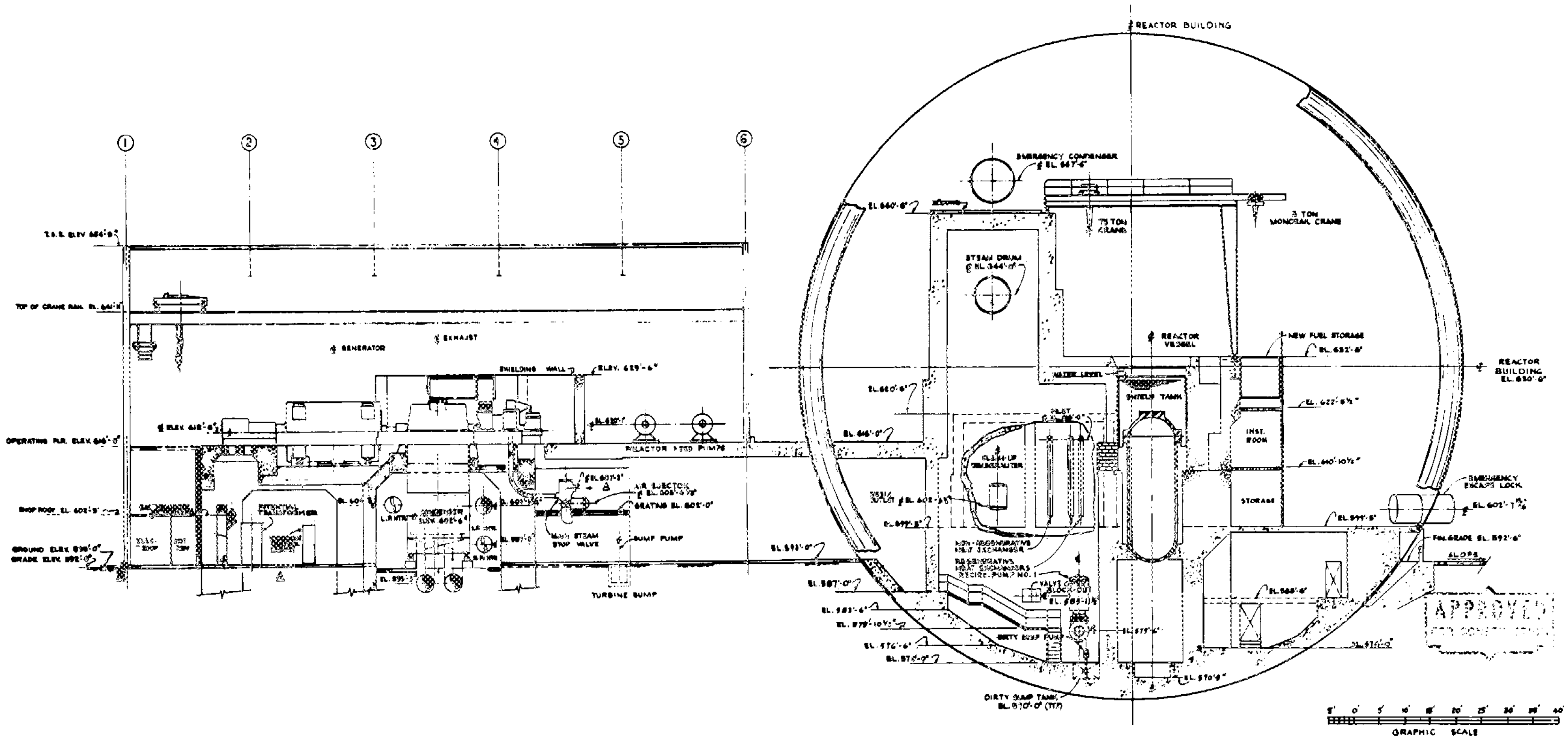



Fig. 7.10. Containment Vessel of the Big Rock Point Plant. (From ref. 8)

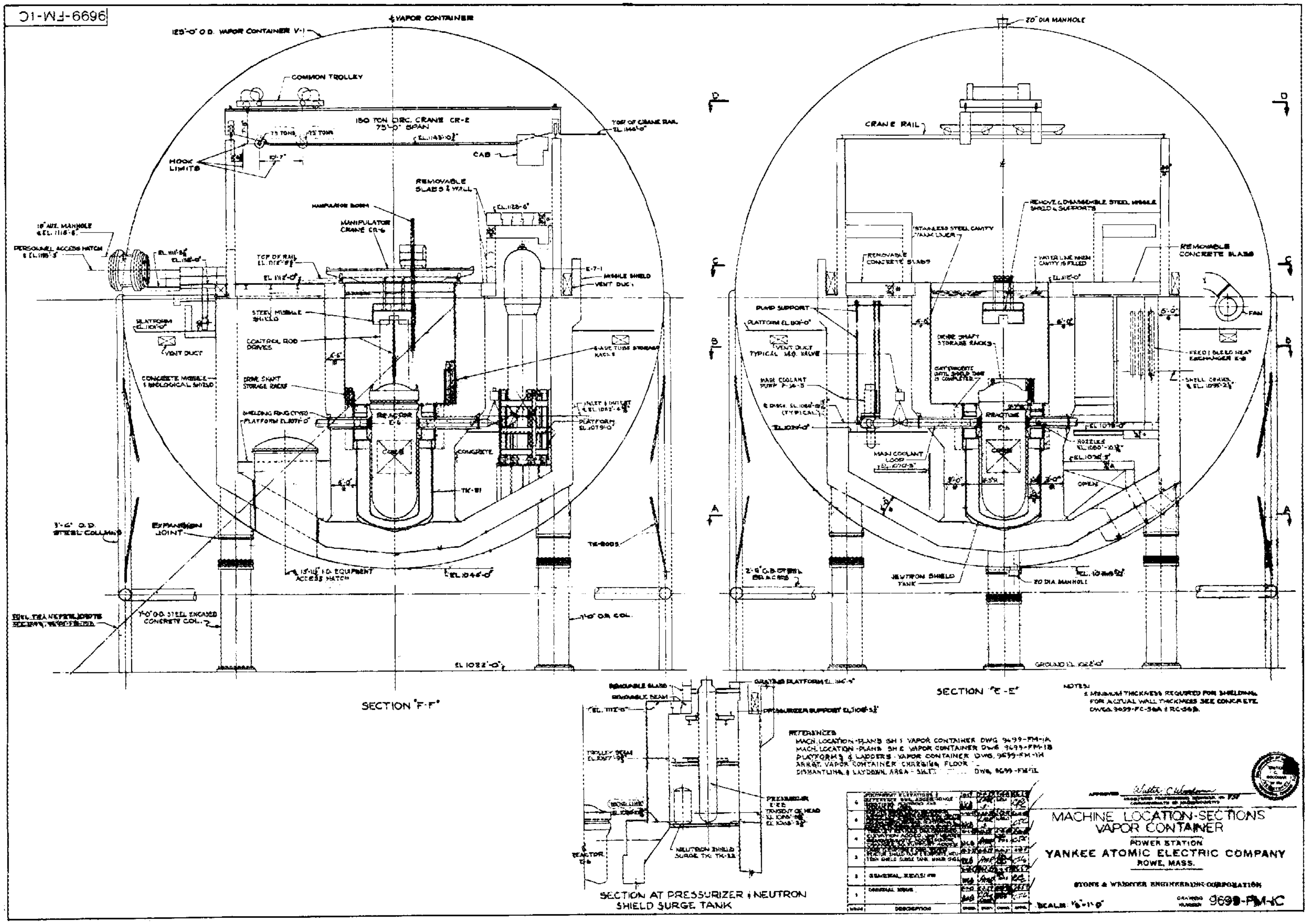



Fig. 7.11. Containment Vessel of the Yankee Atomic Electric Company Plant at Rowe, Mass. (From ref. 9)

of 29.5 psig was selected to provide for coolant-loss energy sources. The initial concept was that the difference between the containment design pressure and the test pressure of 37 psig would permit the consideration of other potential energy sources, such as chemical reactions and nuclear excursions. Later, when the mca concept was developed and the design quantities of system energy were known, it was determined that coolant loss under the most energetic "hot standby condition" would produce a pressure of about 21 psig and that the mca pressure from full-power operating conditions would be 19 psig.[10]

## 7.2.2 Cylindrical Containers

Most cylindrical containers are vertical with hemispherical tops and either hemispherical, hemiellipsoidal, or flat bottoms. This is a large group, and it is further broken down for this discussion by shape of head and construction technique. In general, the cylindrical vessels have a smaller contained volume than the spheres, again with the exception of the EGCR container. The sizes range from a small, 37,000-$ft^3$, free volume in the SM-1 to 450,000 $ft^3$ for the EBR-II and the exceptionally large, 1,360,000-$ft^3$, EGCR containment shell. The design pressures, physical dimensions, and other design parameters are given in Table 7.2.

### 7.2.2.1 Hemispherical Top and Hemiellipsoidal Bottom

Seven vessels comprise a subgroup having hemispherical tops and hemiellipsoidal bottoms (EBR-II, Fig. 7.12; Enrico Fermi, Fig. 7.13; EGCR, Fig. 7.14; Elk River, Fig. 7.15; Pathfinder, Fig. 7.16; PRTR, Fig. 7.17; and Saxton, Fig. 7.18).[11-17] The top is hemispherical because it is the most economical shape, both from the standpoint of amount of steel used and the cost of construction. However, the use of this shape for a bottom would result in much wasted space, since large amounts of concrete would be needed to form the foundations and floors for equipment. The flat bottom would be best from the standpoint of space and equipment arrangement but could present serious stress problems. A compromise between the flat and the hemispherical bottom is the hemiellipsoidal head. This shape relieves the stress problems and less concrete is needed for the floor.

Figure 7.19 shows three containment structures that are the same from the bottom floor up. This figure is intended to illustrate the fact that the hemispherical bottom is the least economical from the standpoint of usable space. It is apparent that much of the space shown as concrete in the containers with spherical and elliptical ends is usable, but even so, the greatest amount of usable space is provided by the flat-bottomed vessel and the least is provided by the spherical-bottomed vessel.

These containers are usually erected with foundations supporting a concrete pad on which the bottom head rests. Concrete is poured inside the building to form the various operating floors, partitions, and foundations. In general, the concrete, both inside and outside, is poured in contact with the vapor container, which is protected from corrosion

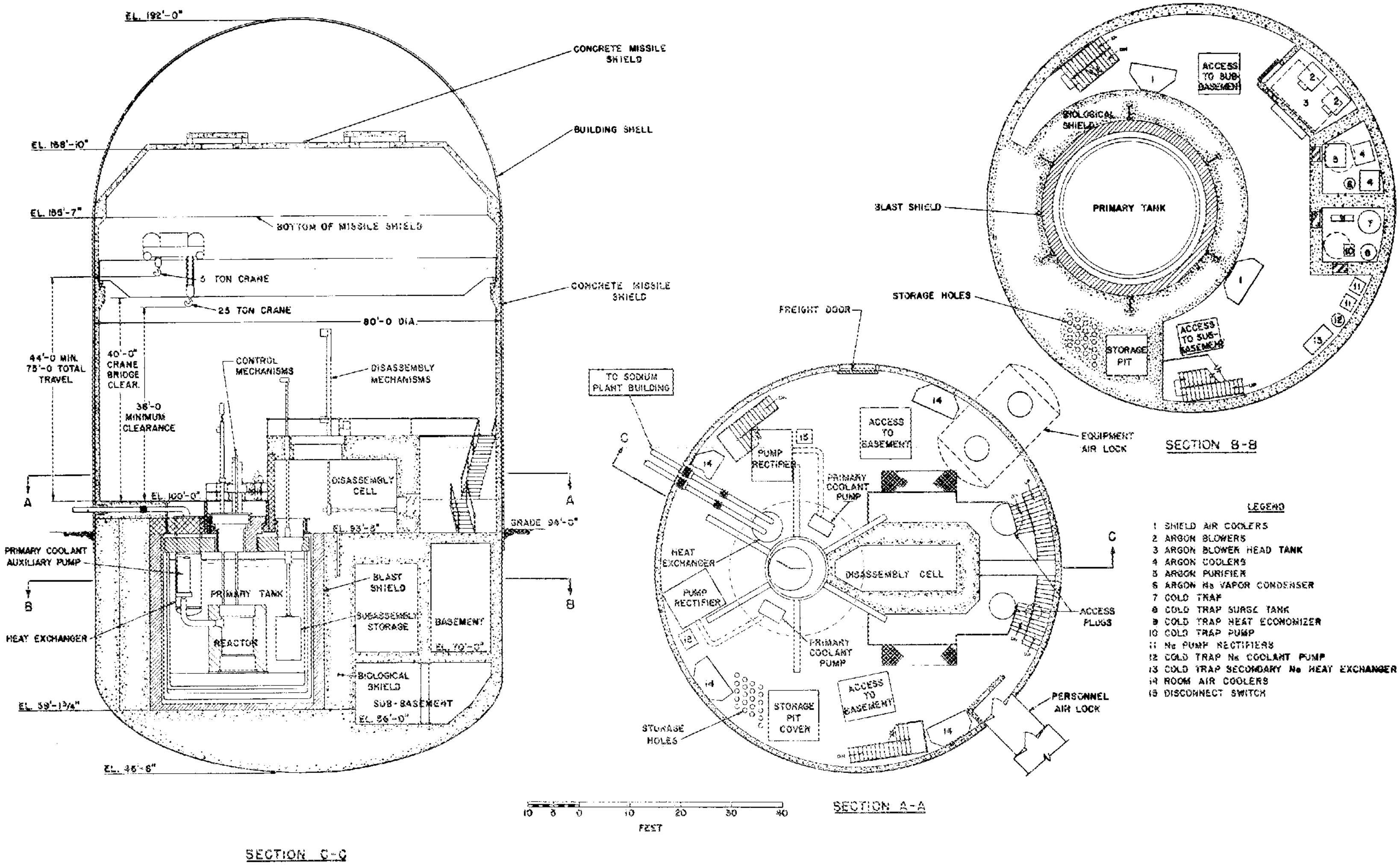



Fig. 7.12. Experimental Breeder Reactor II (EBR-II) Containment Vessel. (From ref. 11)

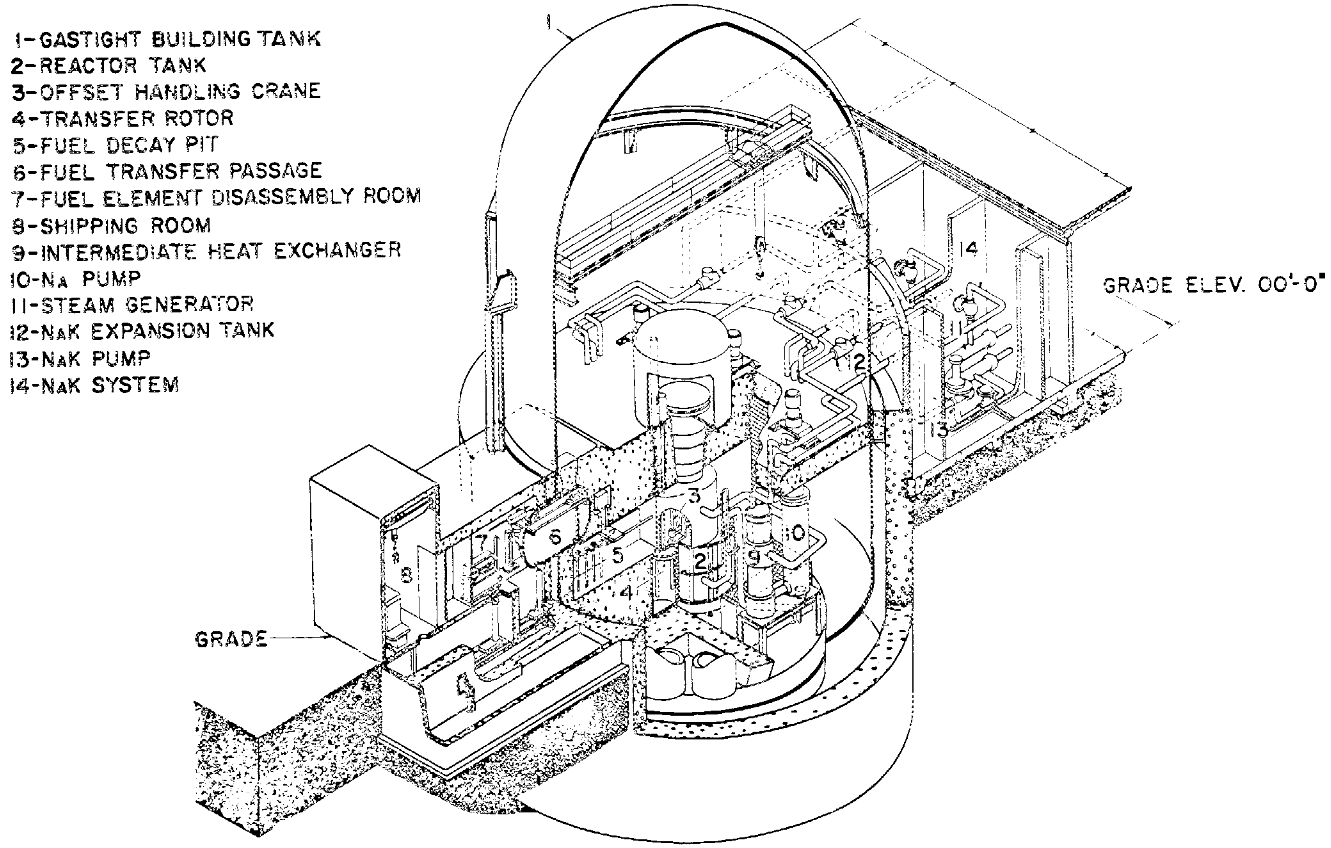



Fig. 7.13. Perspective View of the Enrico Fermi Plant. (From ref. 12)

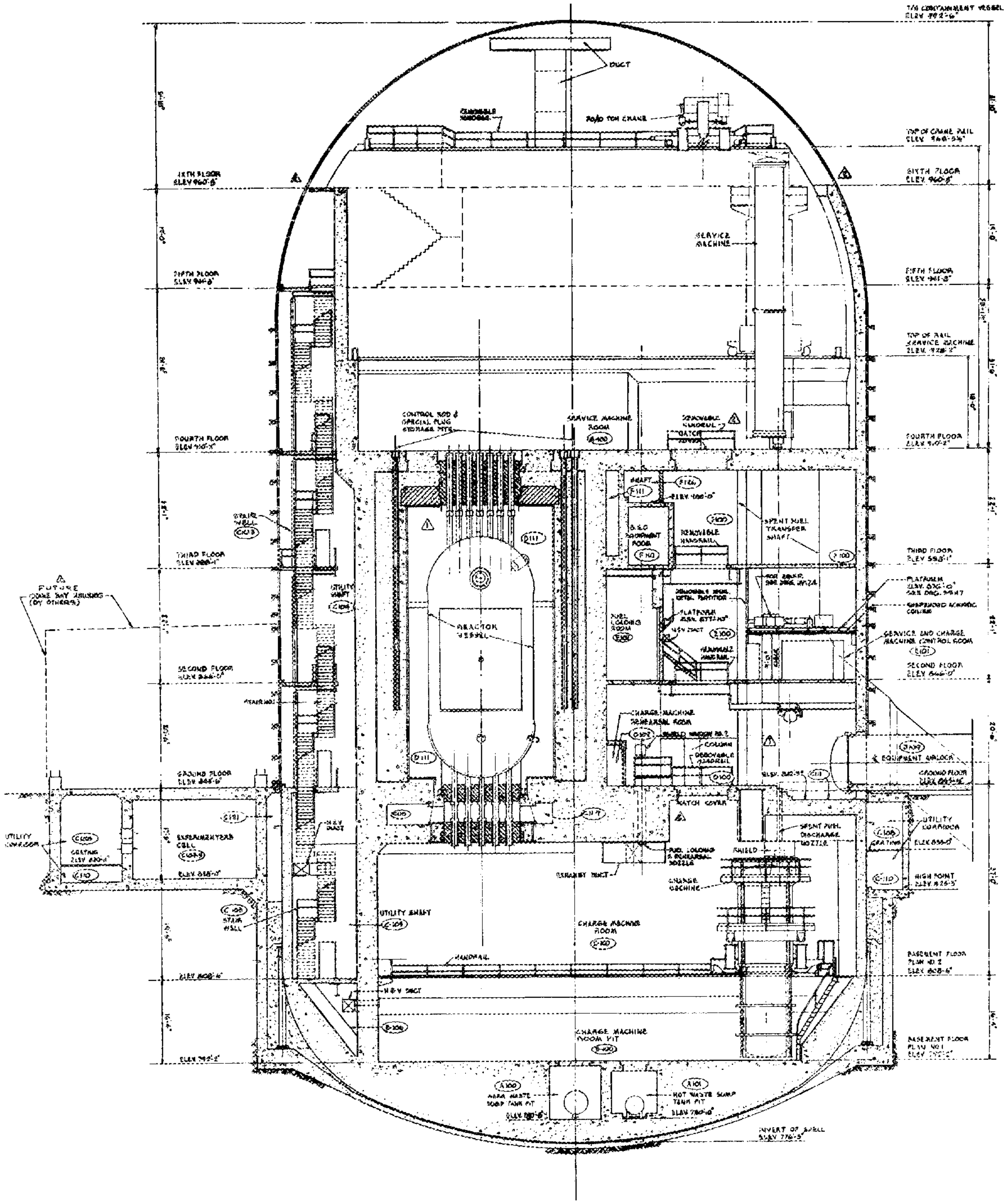
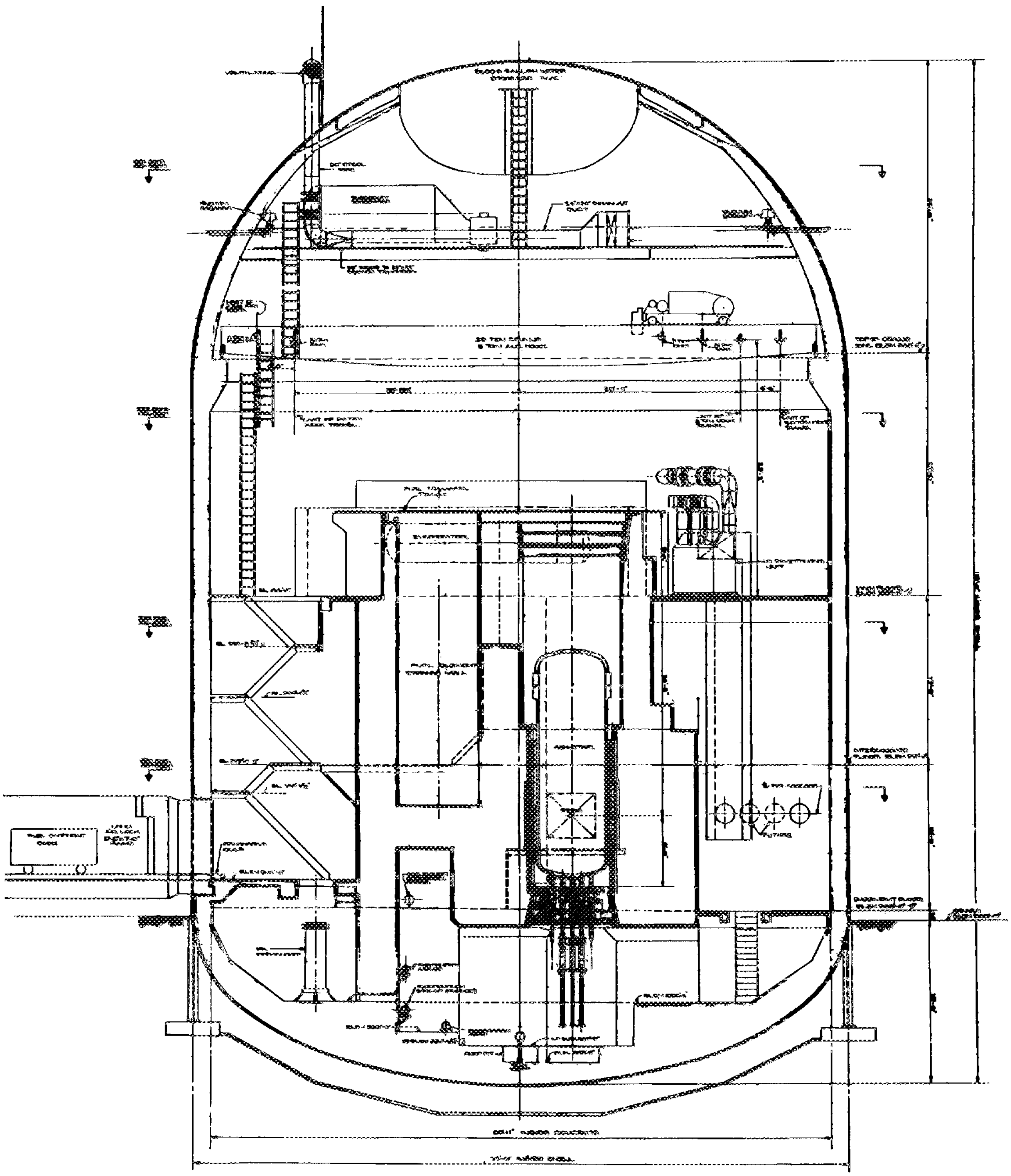


Fig. 7.14. Experimental Gas Cooled Reactor Containment Vessel. (From ref. 13)



GENERAL CROSS SECTION 'A-A'

Fig. 7.15. Rural Cooperative Power Association's Elk River Plant Containment Vessel. (From ref. 14)

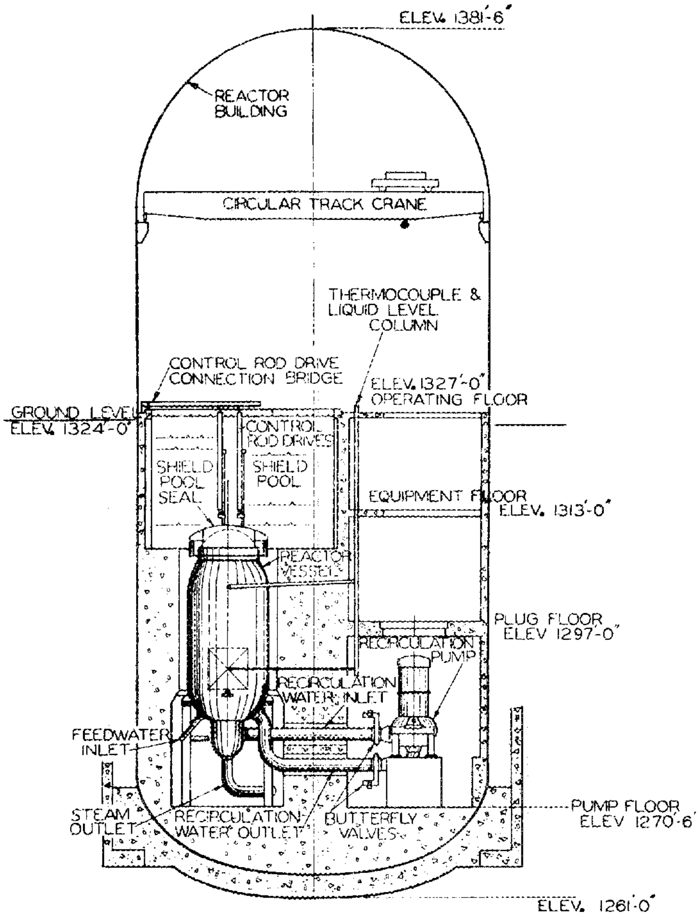



Fig. 7.16. Pathfinder Reactor Building. (From ref. 15)

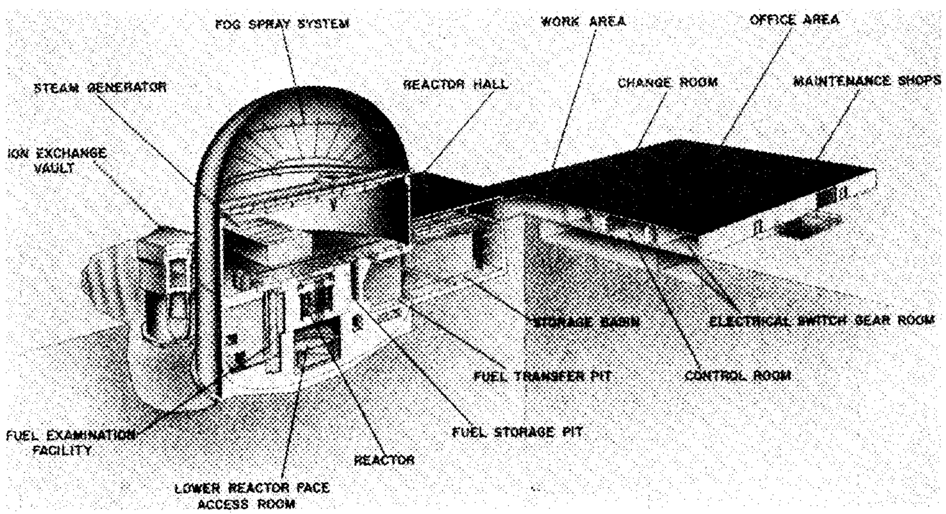



Fig. 7.17. Plutonium Recycle Test Reactor (PRTR) Building. (From ref. 16)

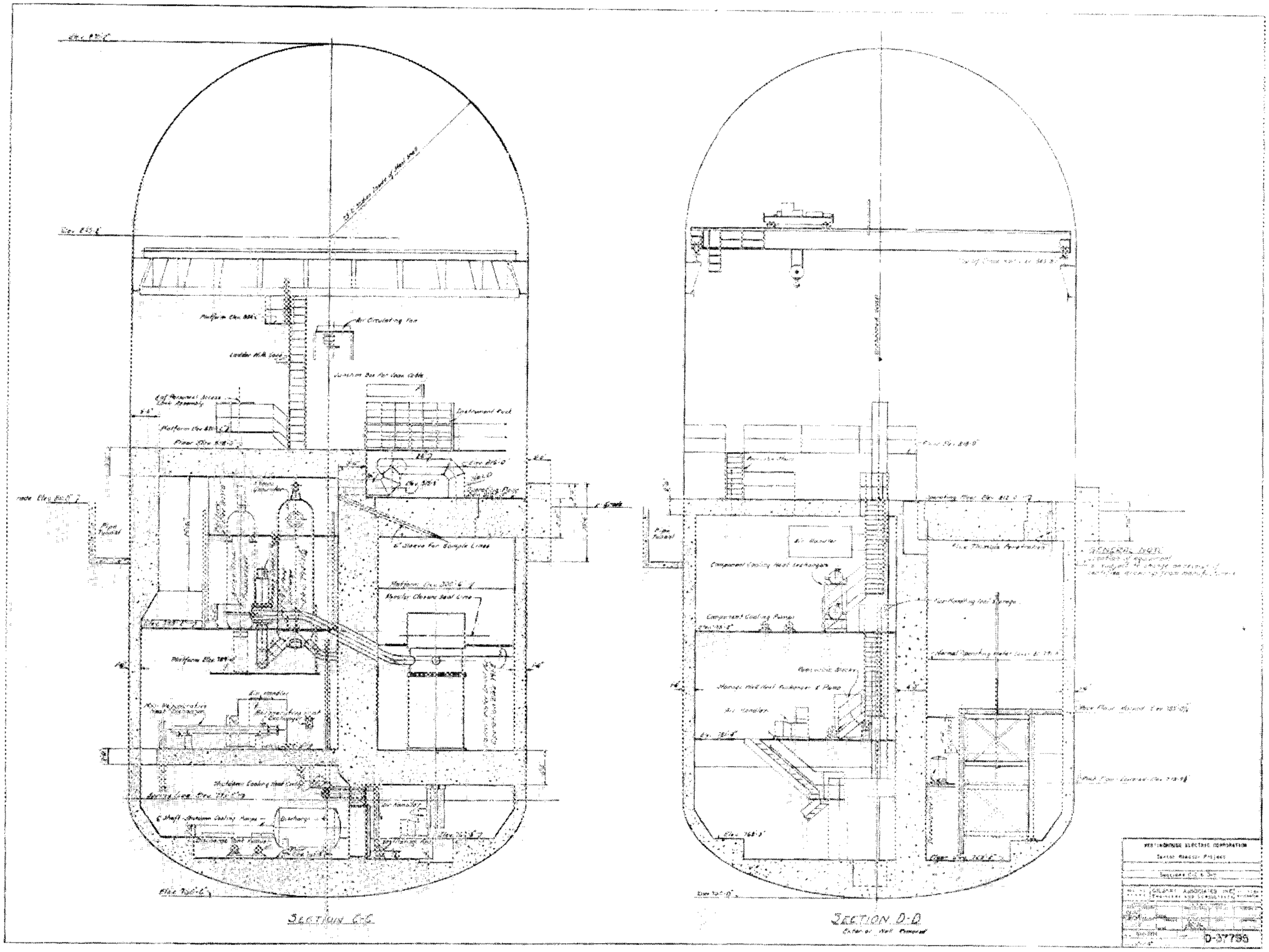



Fig. 7.18 The Saxton Containment Building. (From ref. 17)

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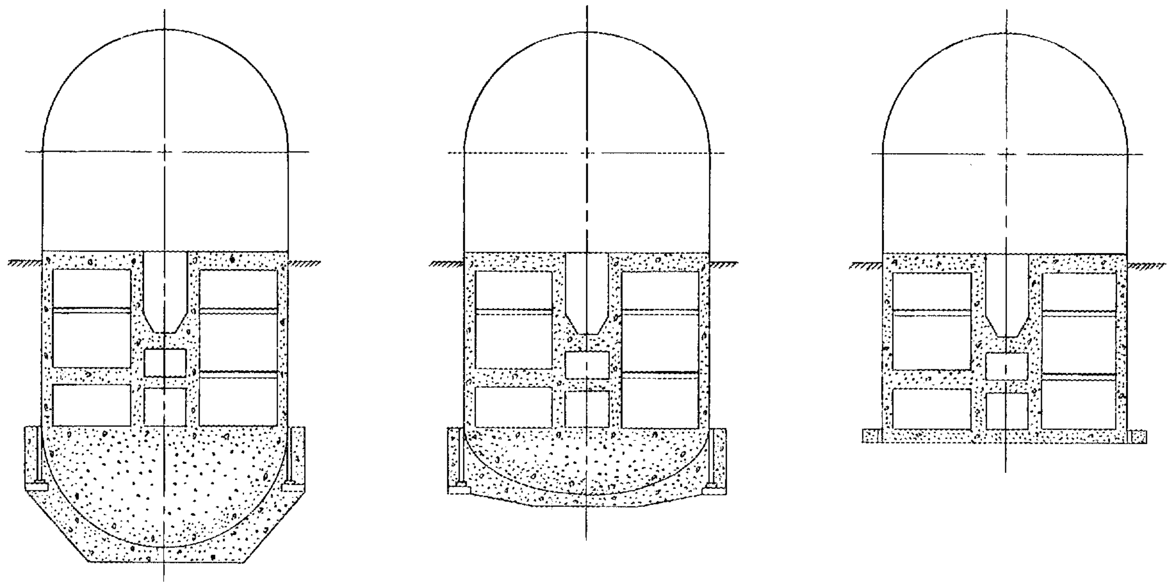



Fig. 7.19. Comparison of Containers with Various Bottom Shapes.

by moisture-resistant paint. In cases where the concrete and steel will not be isothermal at all times, provision must be made to allow for the differences in thermal expansion of the two materials. (This is discussed more fully in Chapter 8.)

The finished grade of the Elk River reactor is approximately at the bend line of the lower head (Fig. 7.15). This puts the main operating floor and the reactor above grade. The reason for this was economic; it was determined that the excavation would have cost more than the additional concrete used for shielding. The EGCR container is also largely above ground, with the reactor vessel completely above finished grade. The reason for this was that the Melton Hill Lake water table would have presented serious problems for deeper excavations. The other five members of this group have their finished grades approximately even with the main floors; this places the bulk of the heavy concrete work and the reactor below grade, allowing the earth to act as shielding from direct radiation.

In all cases the hydraulic and bouyancy effects of ground water must be considered. If this presents a problem, the foundations must be drained, as for the Enrico Fermi Reactor, or the problem eliminated in some other manner.

### 7.2.2.2 Hemispherical Top and Bottom

The VBWR (Fig. 7.20) and the SM-1 (Fig. 7.21) containers[18,19] differ from the foregoing group only in the shape of the bottom. They have hemispherical bottoms that are supported in the usual manner, and the interior

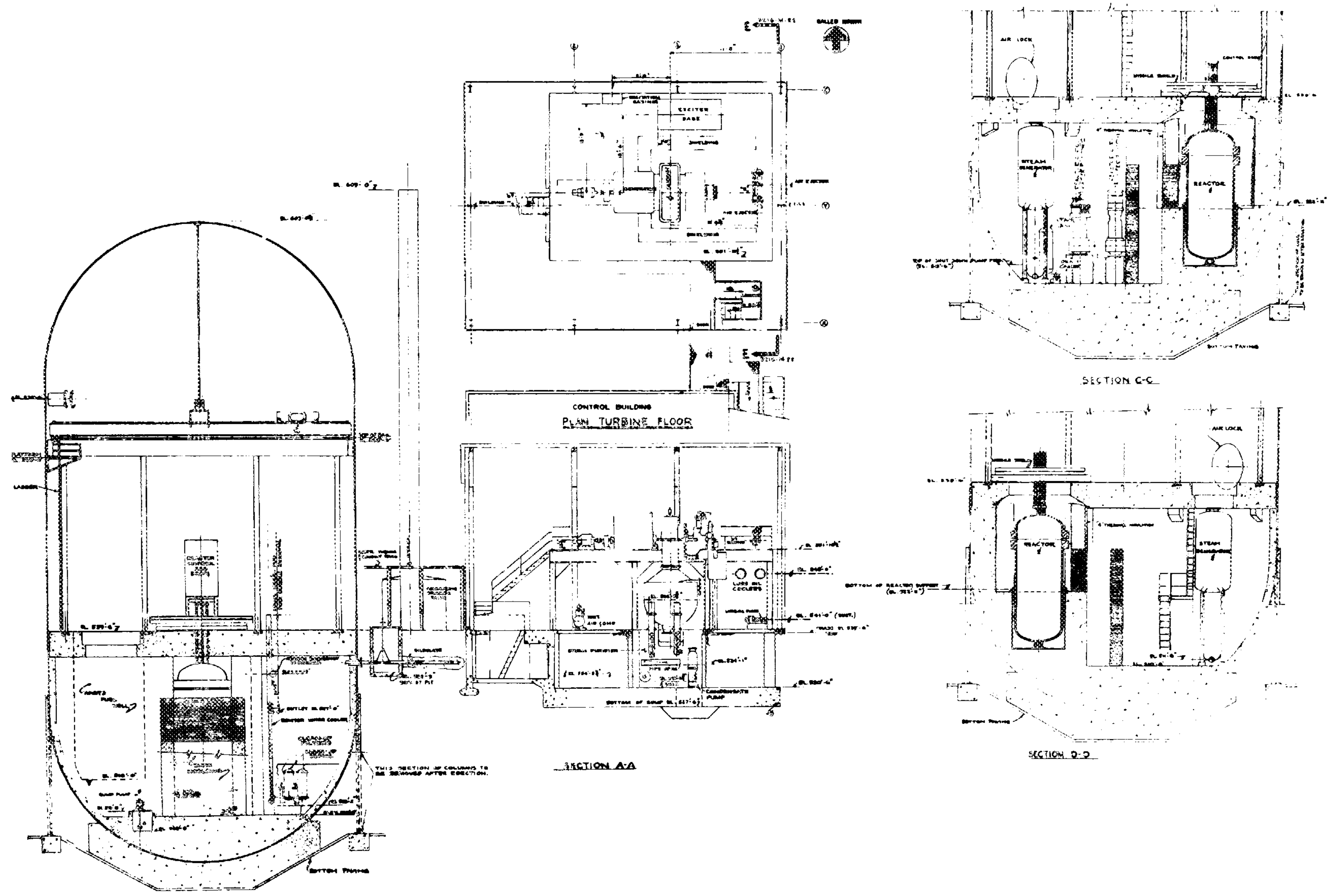



Fig. 7.20. Vallecitos Boiling Water Reactor Containment Vessel.
(From ref. 18)

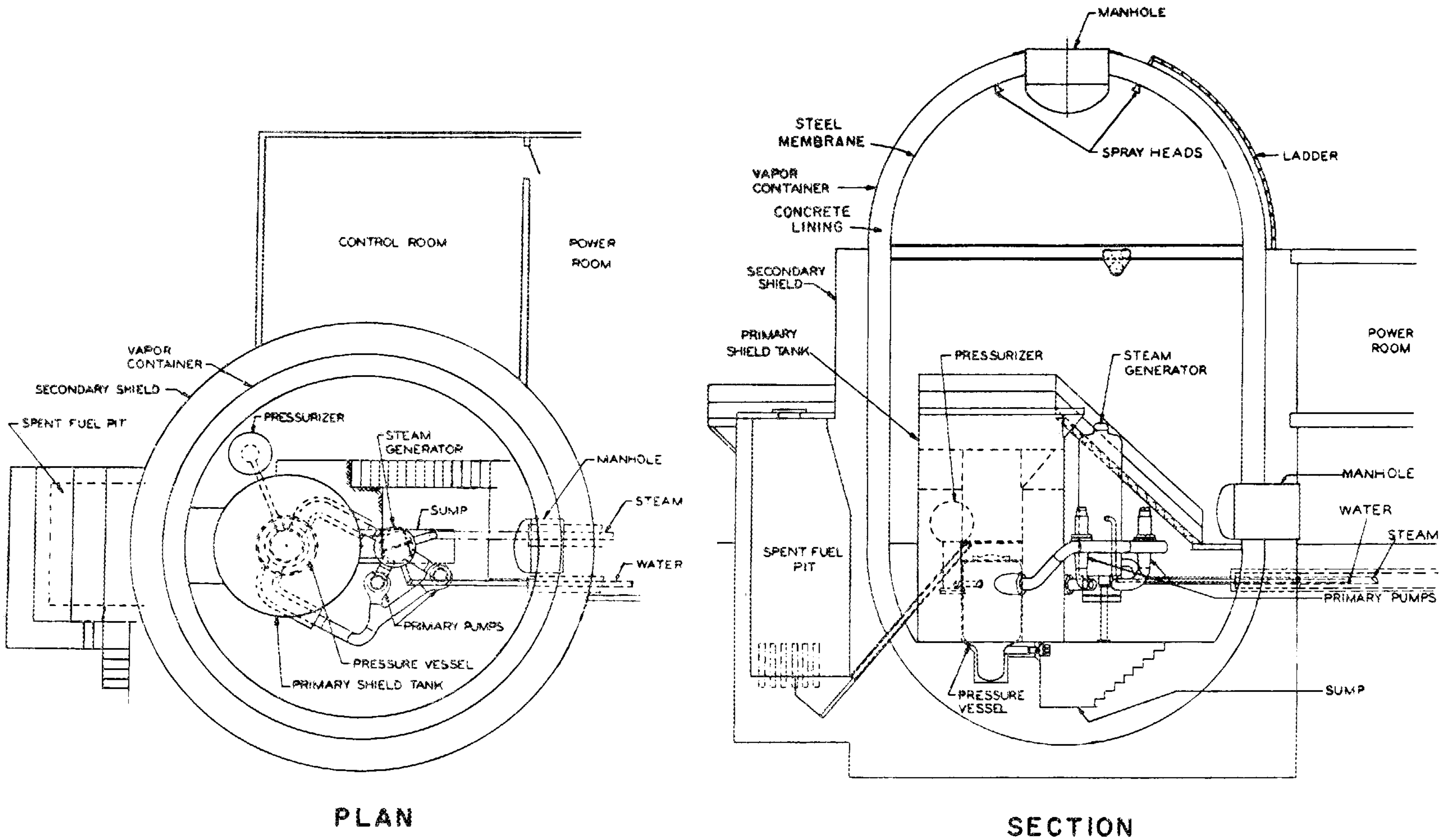



Fig. 7.21. SM-1 Vapor Container. (From ref. 19)

arrangements are similar to those of the other containers, with grade level coincident with the main floor.

Difficulty was experienced in one phase of the construction of the SM-1 vapor container. It was part of the design to pour 2 ft of concrete on the inside and 3 ft of concrete on the outside of the steel containment shell. This concrete was to provide missile protection, as well as shielding. The inside of the inner concrete liner was to be covered with a steel membrane to facilitate cleanup in case of an accident that might spread contaminants inside the container. Erection of this 1/8-in.-thick membrane was difficult, and it is thought that this would not be included in future plants.

### 7.2.2.3 Hemispherical Top and Flat Bottom

1. CVTR. The CVTR container (Fig. 7.22, ref. 20) is a concrete cylinder with a hemispherical steel top and a flat concrete bottom. The materials and method of construction make this system unique. The container was built on a 5-ft 9-in.-thick reinforced concrete foundation pad, and the container itself is a reinforced concrete cylinder with 2-ft-thick walls. The cylindrical portion is 83 ft high and is covered by a 1/2-in.-thick steel hemispherical dome that is covered by 20 1/2 in. of concrete. The top of the foundation mat and the inside of the vertical cylinder wall are lined with 1/4-in.-thick steel plates to form a vapor-tight membrane. The use of this thin membrane is similar in concept to that of the SM-1, with the main difference being the plate thickness. Although the 1/8-in. plate was quite difficult to handle during erection of the SM-1, it proved to be useful when the container was contaminated by a minor activity release. In order to obtain the convenience of the metal liner and yet employ less expensive construction techniques, the thicker 1/4-in. plate was used. Continuous metal battens (welding strips) were cast in the concrete shell. When the concrete work was finished, the steel plates were welded to these battens (see Sec. 7.2.4.3, Fig. 7.28, detail B). All the strength necessary to contain the maximum pressure is provided by the 2 ft of reinforced concrete; no credit is taken for the restraining potential of the steel liner. It was felt that no special attention needed to be given to ambient and seasonal temperature changes. The concrete on the outside of the steel liner is considered to be sufficient thermal insulation that thermally induced stresses will not present any problem.

All internal partitions and equipment foundations are supported by the large foundation mat. Finished grade is about the same level as the main operating floor.

2. HWCTR. The HWCTR containment vessel (Fig. 7.23, ref. 21) has the same general shape as that of the CVTR, but again the method of construction was unique. This container is a composite steel and prestressed concrete structure. The lower half of the building is below grade and is of prestressed concrete 18 in. thick. It is supported by the foundation slab, which is nominally 5 ft thick. The upper half of the building is a conventional steel cylinder with a hemispherical dome.

By using prestressed concrete instead of ordinary reinforced concrete, the required wall thickness was reduced, and the permeability by gas was

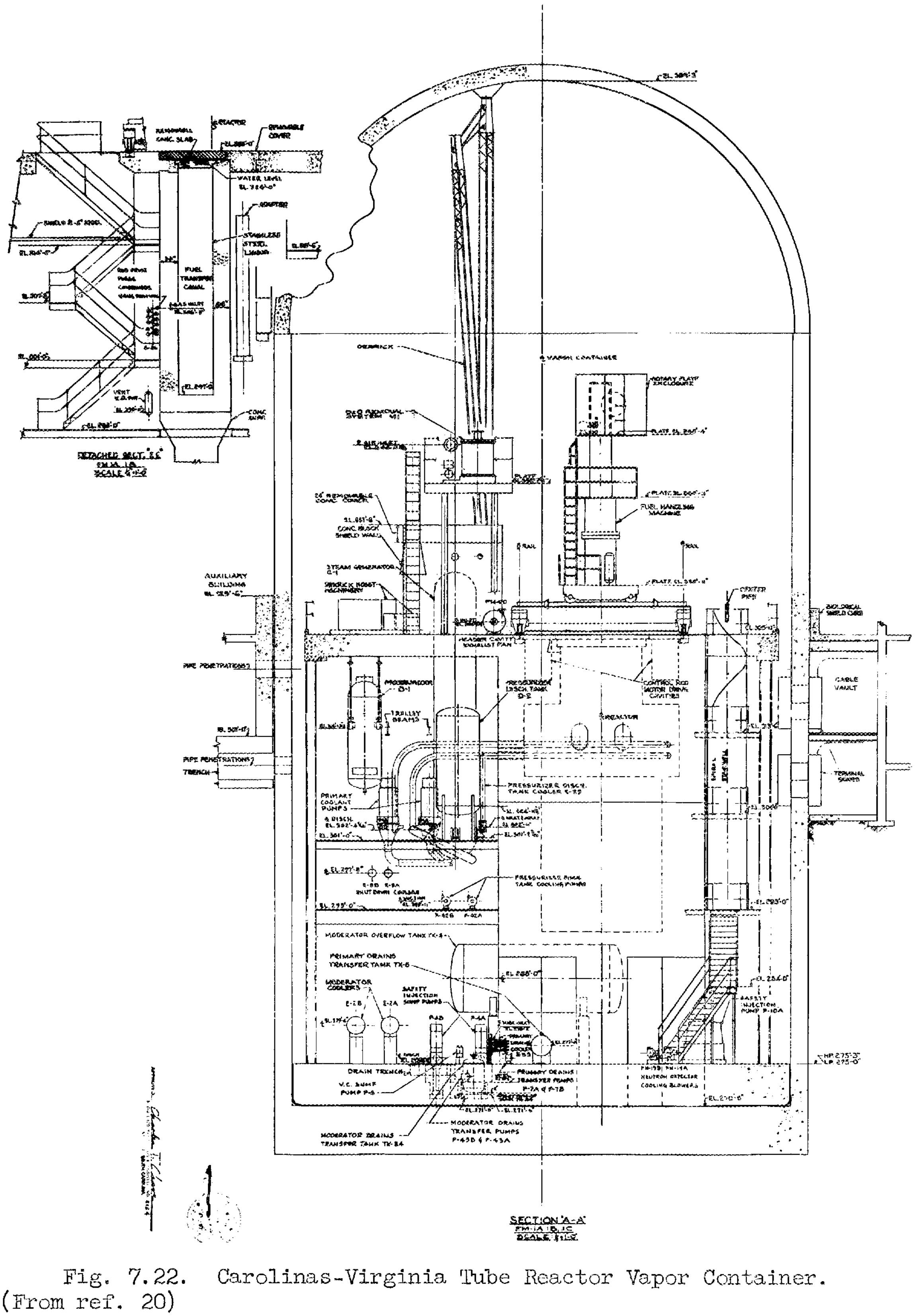


Fig. 7.22. Carolinas-Virginia Tube Reactor Vapor Container.
(From ref. 20)

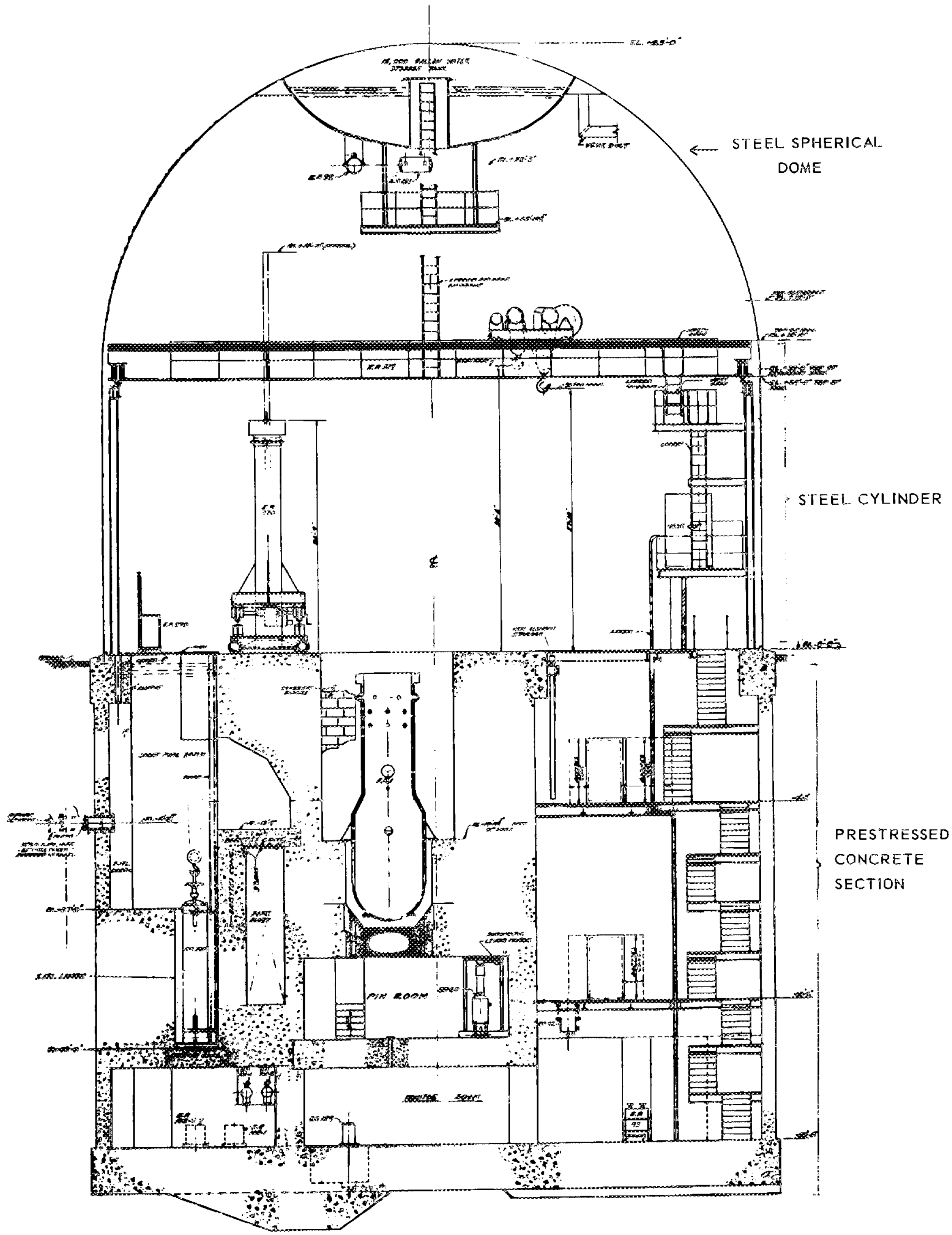



Fig. 7.23. Heavy-Water Components Test Reactor (HWCTR) Containment Vessel. (From ref. 21)

also reduced. Since concrete is porous, the walls that were not covered by steel were coated by an epoxy-base paint to provide a hard nonporous surface for easy decontamination. It is estimated that this container was constructed at a cost of about two-thirds that of a conventional welded steel building.

Points of particular concern were the transition from prestressed to ordinary reinforced concrete and from reinforced concrete to the steel of the upper portion of the container. These transitions are shown in Fig. 7.24. Water stop gates were used to ensure the leaktightness of the vapor container; further, a thermal-setting plastic resin (Liquid Tile) was used, as shown in the details of Fig. 7.24. After completion of the building, further work was necessary because of cracks in the concrete and inability to achieve the required leakage criterion. Most of the plastic paint was used over interior surfaces and at the junction of the floor slab and the cylinder wall. Fiberglas and resin were used to a distance of 6 ft from this joint on both the floor and walls around the full circumference of the building to assure leaktightness of this joint. Considerable time and effort were expended to meet initial requirements.

The differences in thermal properties of the two materials of construction, concrete and steel, could result in serious stress difficulties at the point of transition. This problem was treated in the following manner. The steel superstructure has as its base a fixed tee section flange that is rigidly attached to the concrete structure by 328 bolts placed about the circumference of the structure and by reinforced concrete poured over this tee section. As may be seen in Fig. 7.24, the thick portion of the tee section is embedded in concrete, but the shell wall (the thinner steel) is allowed a 1/4-in. expansion region on each side. It is considered that this is sufficient allowance for movement and deformation of the steel shell due to effects of ambient temperature changes and wind loadings. In the case of the maximum accident, under the influence of an internal pressure of 24 psig and a temperature of 226°F, the lower flange, and also the shell, may be expected to suffer permanent deformation and, upon subsidence of the accident effects, take a permanent set without failure. This is considered to be acceptable.

### 7.2.3 Other Shapes

#### 7.2.3.1 PWR

The PWR container (Fig. 7.25, ref. 22) provides an example of a novel approach to the design of a large containment vessel. Although not as large as the spherical vessels, it has a greater contained volume than any of the cylindrical vessels, except the EGCR container. Its design pressure is rather high, about 53 psig. (For the design philosophy of the PWR container, see Sec. 7.1.2.)

The system consists of four interconnected vessels (three cylinders, and a sphere), with a gross contained volume of 600,000 ft$^3$. The sphere, which contains the reactor, is 38 ft in diameter, with a cylindrical dome (17 ft in diameter, 20 ft high) on top. Located on each side of the sphere are 47-ft-diam, 97-ft-long, horizontal cylinders with hemispherical

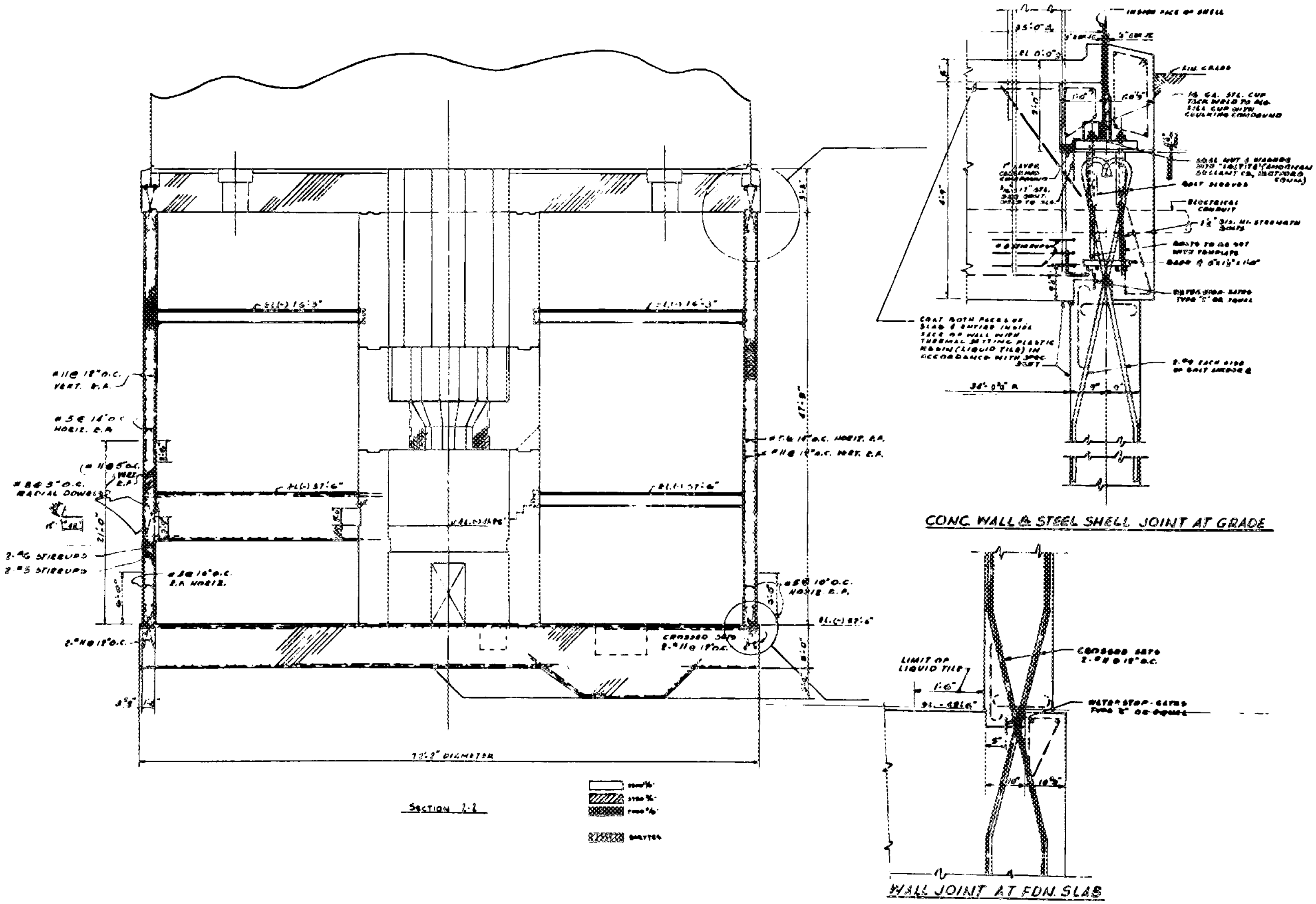



Fig. 7.24. Details of HWCTR Container. (From ref. 21)

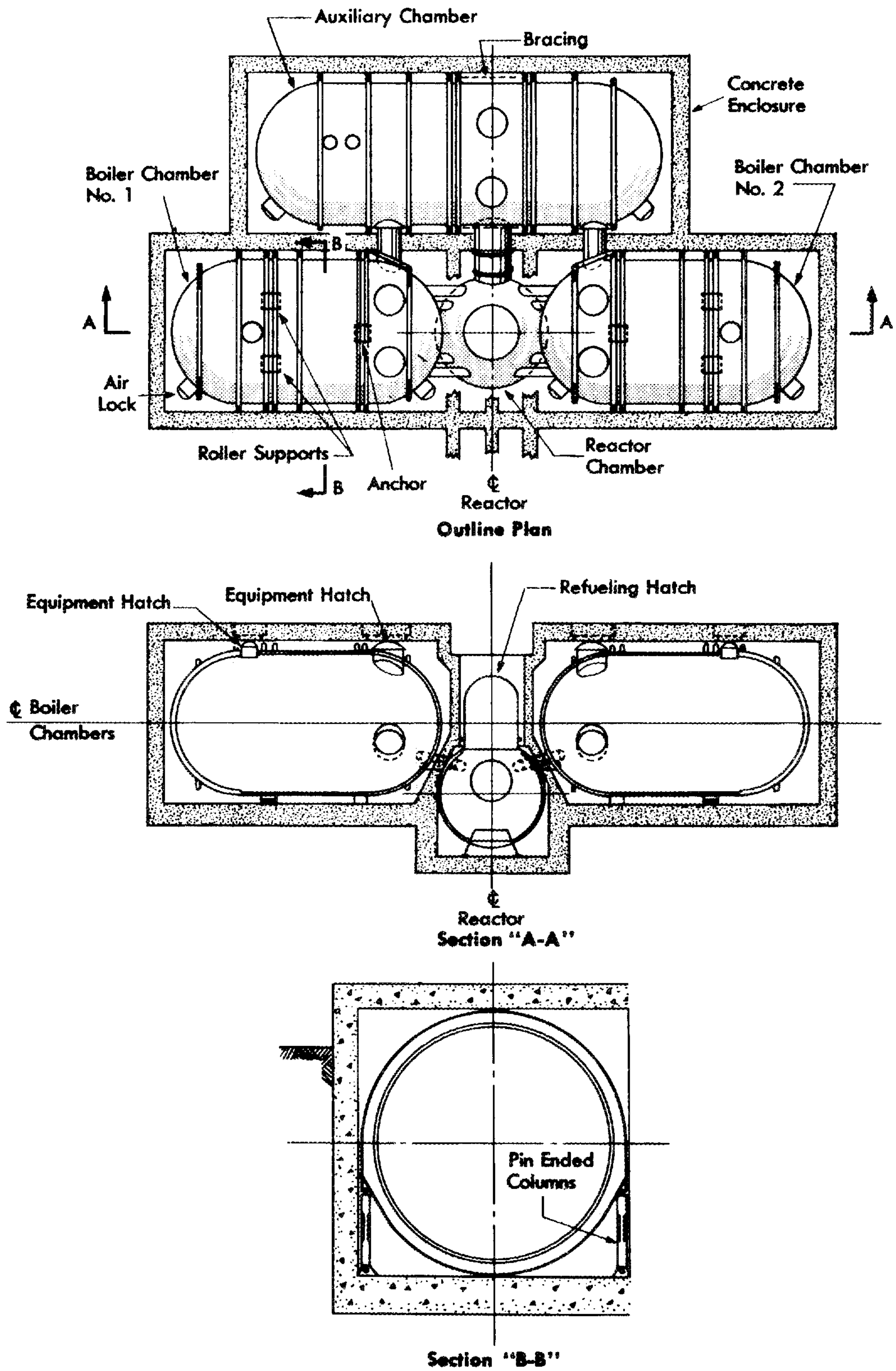



Fig. 7.25. Shippingport Pressurized-Water Reactor Container Arrangement. (From ref. 22)

heads; these are called boiler chambers, and together they contain the four reactor coolant loops and the steam generators. The third cylinder, called the auxiliary chamber, is 47 ft in diameter and 147 ft in length, and it has hemispherical heads. All chambers are interconnected by several large-diameter ducts. The vessels are housed in concrete cells constructed below grade. This type of housing of the vessels permits the periodic inspections required by the Pennsylvania Code.

#### 7.2.3.2 HRT

The HRT container (Fig. 7.26) is a rectangular welded-steel structure erected on a reinforced concrete slab and shielded on the top and sides. The dimensions and shape of the tank were determined by the space available in the building, which had been used for the first homogeneous reactor experiment, HRE-1. The top of the container is flush with the floor, and thus the earth is used as much of the shielding. The roof is made of two layers of removable high-density concrete blocks that are 5 ft in total thickness. The vapor container consists of an all-welded steel membrane liner, nominally 3/4 in. thick on the sides and the bottom and 1/8 in. thick at the top. The top portion of the liner is sandwiched between two layers of shielding blocks and is made up of sections the size of the upper blocks. This makes it possible to remove one or more seal pans for maintenance in a given part of the container without disturbing the entire vapor seal. Further, the below-grade construction makes possible flooding of the container for radiation protection during maintenance. After completion of the container and placement of all penetrations, the inside surfaces were painted with Amercoat 74.

#### 7.2.3.3 NS Savannah

The NS Savannah vapor container (Fig. 7.27, ref. 23) offers an example of high-pressure containment that is unique in several ways. First, it is aboard a ship; second, the design pressure is quite high; and third, the containment vessel is, in a sense, contained. The reactor container is a horizontal cylinder, the axis of which lies fore and aft; it is situated in one of the ship's holds designated as the reactor space. This is located just forward of the bridge and directly below the promenade deck. The reactor space is kept at a slightly negative pressure, and the exhausted air is fed through a filtering system and then up the stack. The design philosophy of this system is similar to that of the Indian Point Reactor, as discussed in Sections 7.2 and 7.8.

The vapor container is, of course, not subjected to loads due to wind or snow or seismic disturbances but may be dynamically loaded by pitching and rolling of the ship. The design criteria define a maximum loading (static plus dynamic) of 0.6 times gravity ($0.6 \times g$). It is estimated, however, that all components can stand at least 1-g loading and in most cases 2 g. These adverse movements are minimized by stabilizing fins protruding from the ship's hull; but, if these were inoperative, the container would withstand the stresses imposed by extreme

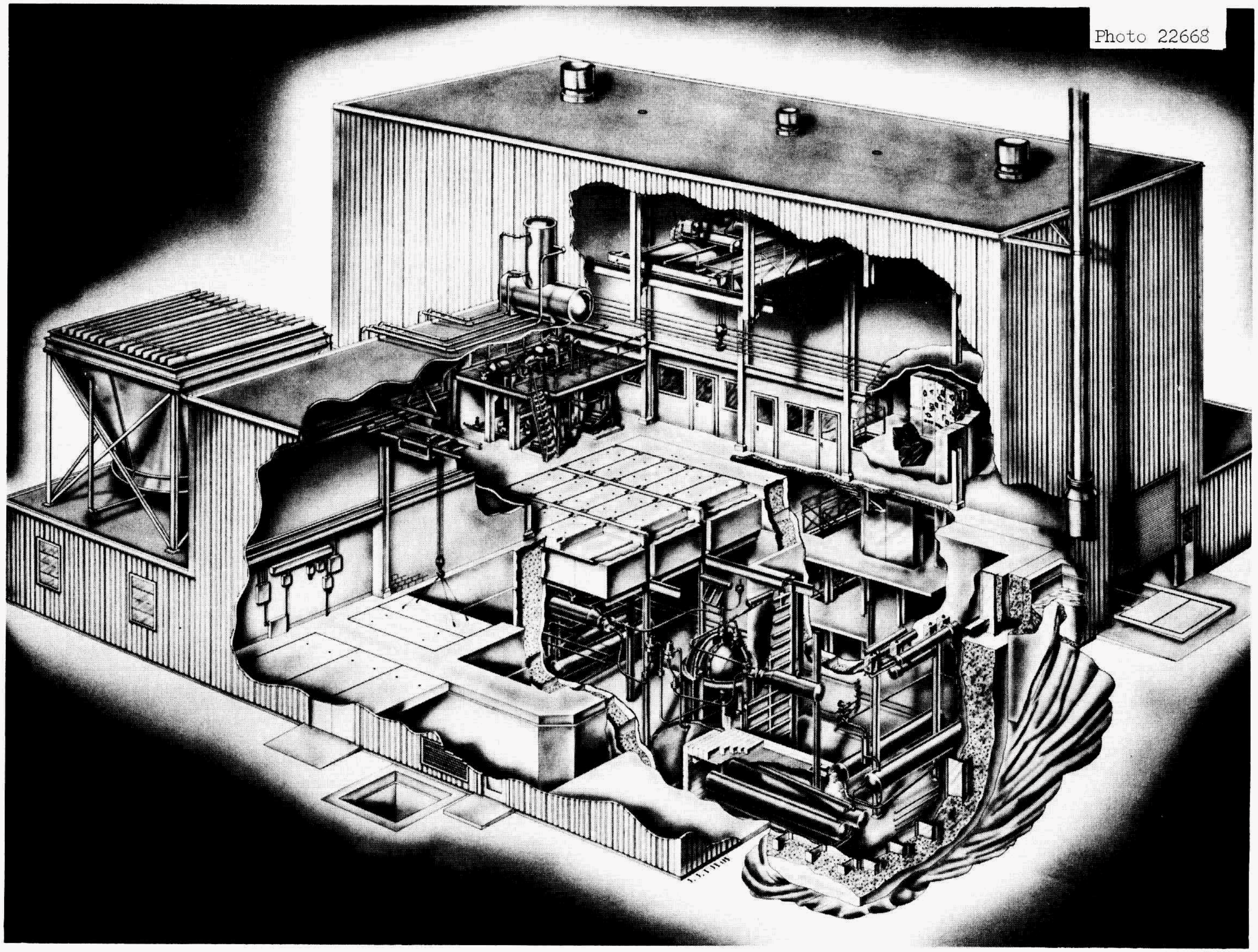



Fig. 7.26. HRT Containment Cell.

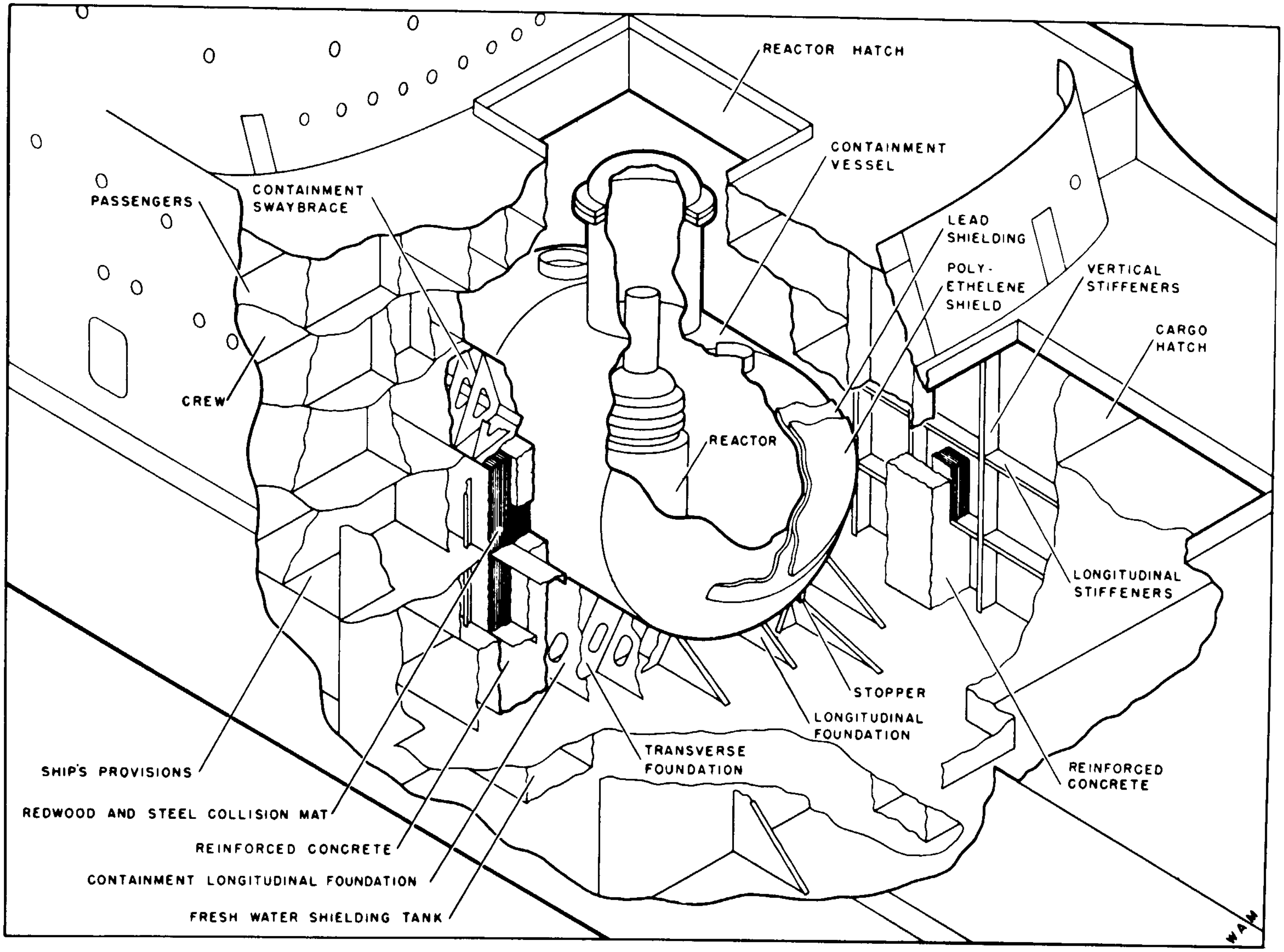



Fig. 7.27. NS Savannah Containment System. (From ref. 23)

pitching and rolling and even those imposed by capsizing the ship. The latter is an unexpected occurrence. Composite redwood and steel collision mats are provided to protect the reactor and containment vessels from damage in the event of collision.

## 7.2.4 Proof Tests

Various tests for the purpose of finding leaks, proving the strength of a vessel, and establishing the leakage rate under certain pressure conditions are performed on vapor containers. Table 7.3 shows test pressures and leakage limitations imposed by the designers and, where possible, the results of their tests.

The methods of measuring the rate of leakage from a vessel are usually the reference-vessel method or the absolute temperature-pressure method. It can be seen from Table 7.3 that the reference-vessel method is used in the majority of cases (see Chap. 10). Most of the containers listed in Table 7.3 were tested at the design pressure for several days to obtain the data for the calculation of a rate of leakage. Some of the reactors are incomplete at this time, and leakage-rate results have not been reported for some others.

In at least five of the cases listed in Table 7.3 (EBWR, Saxton, Elk River, Big Rock Point, and CVTR), a pattern for the tests was established. Upon completion of the containment shell, the testing routine was initiated. First, low-pressure tests were performed at internal pressures ranging from 2 to 10 psig. The container was pressurized to this low pressure, and then a soap solution was painted on all the welds and penetrations. If bubbles appeared or if flaking of the soap occurred, it was obvious that a leak had been found. When all leaks found in this manner were repaired, a second low-pressure soap check was made, and if no further leaks were found, the system was pressurized in steps to 1 1/4 times the design pressure (ASME Code criterion). After pressurizing to 1 1/4 times the design pressure and holding for a specified time (from 15 min to 6 hr), the pressure was then decreased to design pressure and held there for a leakage-rate test of several days' duration. (Methods of testing are discussed in Chap. 10.)

Tests of other systems accomplished the same end but were made in different sequences. In three cases (SM-1, HRT, and NS Savannah) the container proof test of 1 1/4 times design pressure was not required. These were all regarded as special applications of nuclear containment, and the design specifications did not require that the container meet the ASME Code criterion.

### 7.2.4.1 NS Savannah

The NS Savannah leakage-rate test was performed at pressures up to 60 psig, the average 24-hr accident pressure. The long-term leakage rate is the quantity that is of greatest interest; the design pressure would last only a few moments under the conditions of the maximum accident (see Fig. 7.32 in Sect. 7.2.9.2). The average pressure for the next 24 hr gives the most realistic numbers as far as leakage of radioactive contaminants is concerned (see also Chap. 10).

Table 7.3. Containment Structure Proof Tests

| Reactor | Design Leakage Requirements | Low-Pressure Test | Strength Test Pressure (psig) | Leakage-Rate Test Pressure (psig) | Measured Leakage Rate[a] | Method of Measurement | Retest | Frequency of Retest |
|---|---|---|---|---|---|---|---|---|
| Big Rock Point | 0.5%/day | 5-psig soap test | 33.7 (~1 1/4 × dp)[b] | 27<br>10 | 0.036%/day<br>0.021%/day | Reference vessel | At 10 psig | Every 2 years |
| CVTR | 0.1%/day | 1% Freon (sniff) and 5-psig soap check | 26 1/4 (1 1/4 × dp) | 22 1/2 | 0.074%/day (5 days) | Reference vessel | Constant | |
| Dresden | 0.5%/day | 5- to 10-psig soap check | 37 (~1 1/4 × dp) | 29.5<br>27.5<br>5 | <0.1%/day<br>0.0187 ± 0.014%/day<br>0.06%/day | Reference vessel | | |
| Elk River | 0.1%/day | 2-psig soap check | 26.25 (1 1/4 × dp) | 21.5 (two) | 0.09%/day | Reference vessel | | |
| Enrico Fermi | 0.15%/day | 2 psig | 40 (1 1/4 × dp) | 30 | 0.036%/day | Reference vessel | Subsequent test at 2 psig | Yearly |
| EBR-II | 1000 $ft^3$/day at 20 psig | | 30 (1 1/4 × dp) | | | | | |
| EGCR | <0.5%/day | | 11.25 (1 1/4 × dp) | 9 | <0.1%/day (4 days) | Absolute temperature and pressure | | |
| HWCTR | <1%/day | Soap check at low pressure | 29 (~1 1/4 × dp) | 24 | 0.56%/day (3 days) | Reference vessel | Subsequent tests at 5 psig | |
| HRT | 10 liters/min (1.9%/day) | After hydraulic pressure test, soap check for leaks | Hydraulic, 26 top, 36 bottom; pneumatic, 15 | Several times at 15–30 and −7.5 | 4.3 liters/min at 30 psig<br>2.5 liters/min at −7.5 psig | Reference vessel and absolute temperature and pressure | Constant test at −7.5 psig | |
| Indian Point | 0.1%/day | Soap check at 5 in. water | 31.25 (1 1/4 × dp) | 25<br>10 | 0.014%/day<br>0.020%/day | Reference vessel | Continuous by temperature, pressure, and humidity measurements | |
| NS Savannah | 1.5%/day (2.5%/day for an interim period of initial operation) | After proof test, 5-psig soap check | 173 (hydraulic test) | Up to 60 | See Table 10.4 | Reference vessel | | |
| Pathfinder | 0.2%/day | Soap check at low pressure | 97.5 (1 1/4 × dp) | 78 (soap check) | <0.1%/day (no penetrations installed) | Reference vessel | | |
| PRTR | 1000 $ft^3$/day (0.25%/day) | | 18.75 (1 1/4 × dp) | 15 | <1000 scfd (72 hr) | | | |
| PWR | 0.25%/day | Soap test and Freon | 70 (<1 1/4 × dp) | 52.8 | 0.15%/day | Reference vessel | | |
| Saxton | Not stated | 5-psig soap check of all welds and penetrations | 37.5 (1 1/4 × dp) | 30 | 0.04 ± 0.093%/day | | | |
| SM-1 | 1.8 $ft^3$/day | Final leak test with helium | Hydraulic, 76 top | 30 | 0.033%/day | Reference vessel | Periodic | |
| VBWR | <1%/day | Soap check at low pressure | 56.25 (1 1/4 × dp) | 45<br>26.5<br>12 | Without penetrations, <0.1%/day (<0.1% in 60 hr at 25 psig)<br>0.02%/day<br>0.01%/day | Reference vessel | Retest at 15–20 psig | |
| Yankee | 0.1%/day | 5-psig soap check | 40 (<1 1/4 × dp) | 15<br>16 | 0.027%/day at 15.5 psig<br>0.021%/day | Reference vessel | Constant | |

[a]In general these rates were obtained in construction tests after considerable preparatory work had been completed; once a plant has been placed in operation, leak tests should meet the design leakage requirement with no special preparations.

[b]dp is design pressure.

### 7.2.4.2 Indian Point

The containment shell for the Indian Point Reactor met its original design specification upon completion of the shell but before installation of equipment. After all equipment was installed, the "ready to operate" plant was leakage-rate tested at 10 psig, and an extrapolation was made to obtain the rate of leakage to be expected at design pressure.

### 7.2.4.3 CVTR

A concrete building lined with steel plate is difficult, if not impossible, to check for leaks by pressurizing in the usual manner. After the liner plate was installed in the CVTR container, the butt-welded joints were covered with inverted channels (see Fig. 7.28, detail B);[20] both sides of the channels were continuously fillet-welded to the liner plates. The space inside the channel was pressurized through bases (see Fig. 7.28, detail B1) with a mixture containing Freon 12. All channel welds were probed with a halogen leak detector. The channels were used because it would be impossible to determine whether any gas escaped through the butt-welded joints of the liner plate and then through the exterior concrete.

### 7.2.4.4 Future Testing (See also Section 10.6)

Most of the reactors do not include in their future operating procedures provisions for periodic or continuous containment leak testing. Many indicate that future leak testing will be conducted if necessary.

At HRT, however, it was possible to conduct continuous leakage-rate surveillance during reactor operations. The HRT cell was maintained at a negative pressure of 7.5 psi during operation so that meaningful leakage rates could be calculated using reference-vessel differential-pressure data.

The Yankee plant has provisions for continuous monitoring of the building leakage. The containment shell is pressurized to 1 psig, and a bank of compressed-air bottles is maintained to keep the shell pressure at this level. Periodic readings are made on the mass of air which has been admitted to the containment over a period of time. It is considered that if the period over which the leakage is calculated is long (several days), then temperature effects will cancel out and a believable leakage rate will be obtained. Since the container is pressurized at all times, the reference-vessel system is used also to determine the leakage rate in a continuous fashion. The CVTR has a similar system. Such systems serve principally to detect development of major leaks during operating periods and may eventually be considered as a substitute for integrated leakage-rate tests at design pressure once the leakage rate versus pressure relationship has been established for a given structure.

The HWCTR containment building is significantly different from most high-pressure containers in that the below-grade prestressed-concrete portion is not steel lined and depends upon the limited porosity and cracking of this type of concrete to maintain its vapor integrity. Although still considered promising, this system has not been without

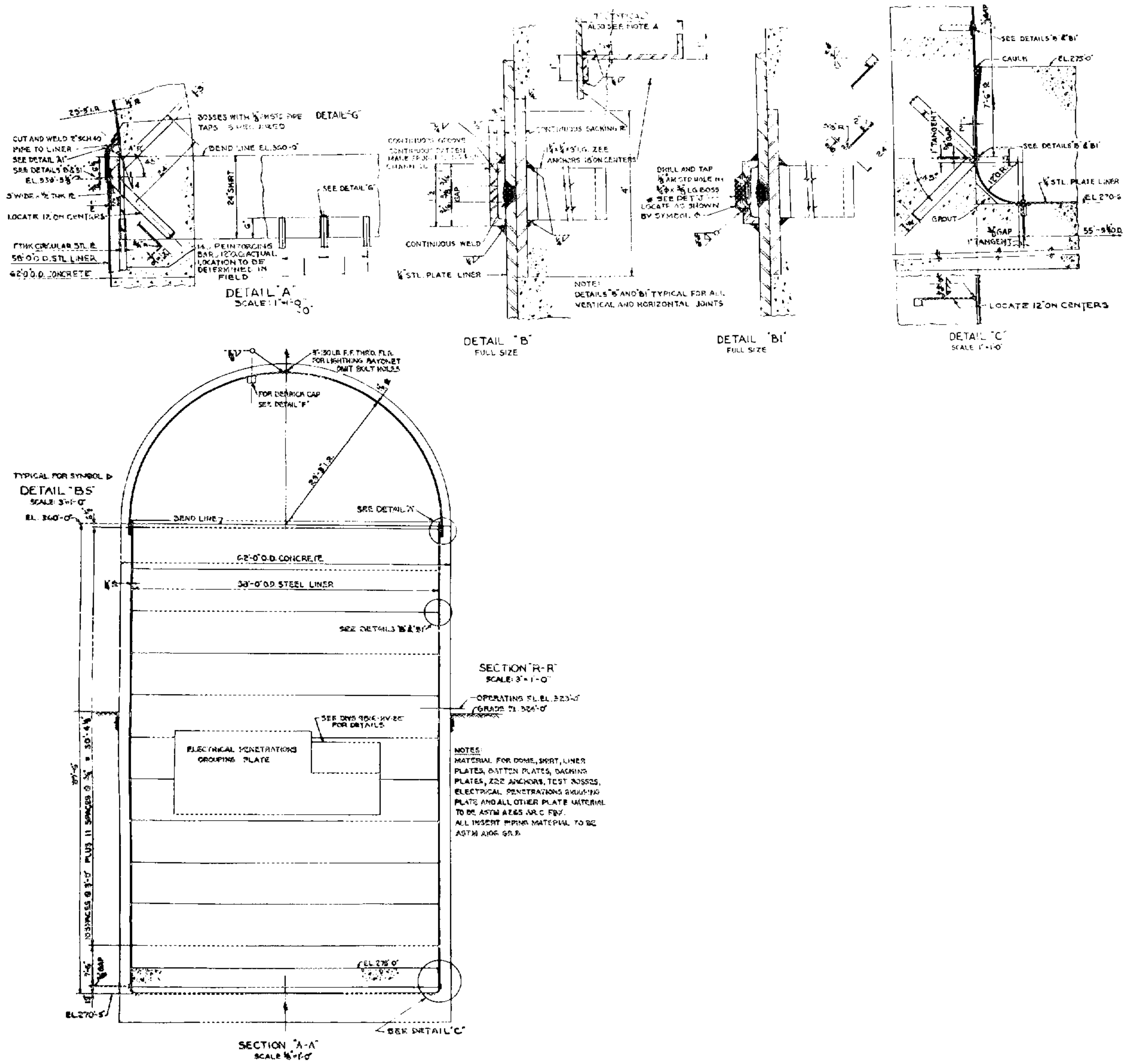


Fig. 7.28. Details of the CVTR Container. (From ref. 20)

serious difficulties. The building met its leakage requirement of <1%/day (0.56%/hr). Then, when equipment was placed in it and the system was made ready to operate (except for installation of pressure-sensitive instruments), a retest was made. This test yielded leakage values well in excess of the allowable rate. By repairing conventional openings, such as a ventilation valve that was not closing tightly, an air-lock gasket that was leaking, and some others, this rate was reduced substantially. Additional repair work on the interior concrete at corners and under support pillars, where several cracks had appeared, was made, and the repairs were coated with Fiberglas and resin. This brought the leakage rate down to the specified range. It was felt by the plant management, however, that some good way must be provided to observe the leakage rate periodically. After installation of pressure-sensitive process equipment, it was not considered practical to retest at the design pressure, since this would entail removal of the sensitive equipment. To circumvent this problem, leakage tests were made at 5 and 12 psig, and these data were extrapolated to the design pressure of 24 psig. As it stands now, the leakage rate will be taken at 5 psig and multiplied by 3.8 to yield the leakage rate at 24 psig.

Similar work has been done by those who are responsible for the NS Savannah, where the retest schedule is more demanding than for an ordinary reactor container. For the HWCTR, retesting at high pressure is considered impractical, so experimental work was done by EBASCO to obtain a family of curves by which the leakage at some higher pressure could be predicted by leakage-rate data at some low pressure. Subsequently, testing of the NS Savannah container at various pressures up to 60 psig provided actual data on the leakage rate versus pressure. More general leakage versus pressure relationships have been derived by Maccary and summarized in a report,[24] which is now being revised.

#### 7.2.4.5 Leakage-Rate Specifications

Above a certain rate the specification of a leakage rate is determined more by the desire to minimize accident exposures than by a calculated need, since a nuclear plant constructed in a desolate area could be allowed to have a much larger leakage rate than one built in a more populated area. It can be seen from Table 7.3, however, that most of the plants have a similar leakage rate specified, irrespective of the location.

Table 7.3 shows that with few exceptions the specified leakage rate ranges between about 0.1 and 0.5% of the contained volume per day. Exceptions are SM-1 and Enrico Fermi, which have leakage rates lower than this, and VBWR, HRT, and NS Savannah, which have higher specifications.

A case in point might be the EGCR. A careful analysis was made of the potential release of fission-product activity from the core, the containment pressure, the worst conceivable meteorological condition, and the site-boundary distance. Using these parameters and the allowable doses at the exclusion boundary, an allowable leakage rate of 2.2%/day was calculated.[25] It was felt by the Commission, however, that for this type of building a much better leakage rate could be attained, and the specification was thus revised to <0.3%/day.[26] The stand that the Commission takes in this sort of case is justifiable, particularly for a new system. A containment system depends upon large, quick-closing valves to ensure leaktightness of ventilation ducts and similar penetrations. When new, these may be expected to be leaktight, but after having been used for several months, their sealing ability may be decreased.

However, in early 1964 the attitude toward the leakage rate was altered slightly; the design leakage is to be met at the time of construction to prove that the fabricator has properly done his job, but subsequent leakage-rate tests are to meet site criteria boundary dose requirements, which may allow a leakage rate higher than the one specified for design purposes.

### 7.2.5 Materials and Specifications

Table 7.4 lists codes, specifications, and types of material used in the containers and the size and material of the reactor primary pressure vessel. In general, the high-pressure containers have been built in conformance with Section VIII (including the related code cases) of the ASME Boiler and Pressure Vessel Code. New Section III on Nuclear Vessels will apply to future containment vessels.

In column 5 of Table 7.4, a nil-ductility transition (NDT) temperature for the containment vessel is listed for four reactors. The importance of this number is sometimes questioned, since the ASME Code acceptability of the given steel covers this. (See Sec. 8.5 for a discussion of the brittle fracture of steel.) The technical specifications of the Elk River Reactor require that the temperature[27] of the steel container shall not be allowed to go below 10°F. In column 6, the 100% x-rayed entry is usually taken to mean that all welds are x-rayed; but, if it is impossible to obtain a meaningful representation on x-ray, magnetic-particle or dye-penetrant tests must be made.

In some cases the ASME Code does not apply. In this event, the "spirit" of the Code is adhered to. The HWCTR was faced with this situation because the lower part of the vessel was concrete and no code specifically applied.

The CVTR method of testing welds is fully described in the previous section. This method was not according to the code, but the code could not be followed in this case.

### 7.2.6 Penetrations

The penetration of the vapor container wall by any means serves to compromise the containment and requires that very careful precautionary measures be taken to ensure closure during an accident. General details of ventilation designs are presented in Chapter 9. Information concerning the number, size, and type of penetrations is given in Table 7.5, and Table 7.6 indicates the number of closing devices per line and the logic used to close these lines in case of accident or emergency conditions.

It may be seen in Table 7.5 that the HRT containment structure is peculiar in that it is penetrated a large number of times. Many things account for this anomaly. There are more than 200 penetrations for the primary system leak-detection lines, more than 100 for the elaborate refrigeration system in the cell, and a host of instrument penetrations

Table 7.4. Codes and Specifications Used in the Fabrication of Containment and Primary Pressure Vessels

| Reactor | Code | Containment Vessel | | | | | Reactor Primary Pressure Vessel | | | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | Steel Specifications | Thickness (in.) | NDT Temperature[a] (°F) | Weld Inspection | Vessel Code Stamped | Inside Diameter (in.) | Wall Thickness (in.) | Type of Metal | NDT Temperature (°F) |
| Big Rock Point | ASME | SA-201-B, SA-300 | 0.701–0.87 | | 100% x-rayed | Yes | 106 | 5 1/4 (clad) | A-302B steel | <10 |
| CVTR | ASME ACI-318 | A-285-C for liner plate | Side, concrete (24) with 1/4-in. steel liner plates<br>Top, 1/2-in. steel with concrete (20 1/2 in.) shielding | | Dye penetrant and halogen leak tested (see discussion) | No | 3.53 (pressure tube) | 0.191 | Zircaloy-4 | |
| Dresden | ASME | A-201-B, A-300 | 1.25 (min)<br>1.40 (max) | −20 | 100% x-rayed | Yes | 146 | 5 1/4 (+3/8 clad) | A-302B steel | 10 |
| Elk River | ASME | A-201-B, A-300 | Side, 0.7<br>Top, 1/2<br>Bottom, 0.7 | −50 | 100% x-rayed | Yes | 84 | 3 | A-302B steel | 100 |
| Enrico Fermi | ASME | A-201-B, A-300 | Side, 1<br>Top, 5/8<br>Bottom, 1 1/4 | | 100% x-rayed | Yes | 170 (upper)<br>111 (lower) | 2–1.5 | A-240-45 steel, 304 SS | |
| EBR-II | | A-201-B | Nominally 1 | | 10% x-rayed | No | 73.75 | | | |
| EGCR | ASME | A-201-B, A-300 | | | ASME Sec. IX; x-rayed | | 240 | 2 3/4 | A-212B steel | |
| HWCTR | ASME | A-201-B, A-300 | Top, 11/32<br>Side, 3/4 (concrete, 18 in.) | | 100% x-rayed | Yes | 86 | 4 1/2 (+1/4 SS[b] clad) | A-212B steel | <110 after 20 yr |
| HRT | | A-285-C | Floor and walls, 3/4<br>Ceiling, 1/2 | | Dye checked | No | 60 | 4 (+0.4 clad) | A-212 steel | |
| Indian Point | ASME | A-201-B, A-300 | Upper, 0.89<br>Lower, 1.03 | −50 | 100% x-rayed | Yes | 117 | 6.94 | A-212B steel (clad: 0.109 in. 304 SS) | 65 (120°F after irradiation) |
| Shielding building | | | Wall, 66<br>Dome, 33 | | | | | | | |
| NS Savannah | ASME USCG | A-212-B | Cylinder, 2 5/8<br>Heads, 1 1/4 | | All but welds x-rayed | No | 98 | 6 1/2 | A-212B | <110 after 20 yr |
| Pathfinder | ASME | SA-212-B, A-300 | Side, 1 3/8<br>Top, 11/16<br>Bottom, 1 3/8 | −50 | | Yes | 132 | 2 3/4 (+1/4 SS clad) | A-212B | |
| PRTR | ASME | A-212B | | | | | 3.250 (pressure tube) | 0.154 | Zircaloy-2 | |
| PWR | ASME | A-201-B, A-300 | 1.25 (max)<br>0.66 (min) | | 100% x-rayed | Yes | 109 | 8 3/8 (+1/4 304L SS clad) | A-302B steel | |
| Saxton | ASME | A-201-B, A-300 | Side, 3/4<br>Top, 11/32<br>Bottom, 11/16 | | | | 58 | 5 (multilayer) | A-212B steel | |
| SM-1 | ASME | A-201 | Top, 1/2<br>Bottom, 1/2<br>Side, 7/8 | | 100% x-rayed | No | 48 | 2.5 (+1/4 clad) | A-212B steel | |
| VBWR | ASME | SA-212-B, A-300 | Side, 7/8 and 1<br>Heads, 7/16 | −11 | 100% x-rayed | Yes | 84 | 3 3/8 | A-212B steel | |
| Yankee | ASME | A-201-B, A-300 | 1 1/4 | | 100% x-rayed | Yes | 109 | 7 7/8 | A-302B steel (clad: 0.109 in. 304 SS) | −10 initially |

[a]These values determined by Charpy V-notch impact tests.

[b]SS = stainless steel.

Table 7.5. Containment Structure Penetrations

| Reactor | Air Locks | | Access Openings and Doors | Other Penetrations | |
|---|---|---|---|---|---|
| | Number | Description | | Number | Description |
| Big Rock Point | 3 | 12 ft diam<br>7 ft 7 in. diam<br>5 1/2 ft diam | | 95 | Electrical and piping |
| CVTR | 0 | No access during operation | 13-ft 1-in. bolted and gasketed equipment door<br>7-ft 1-in. bolted and gasketed personnel and equipment door<br>3 1/2-ft bolted and gasketed emergency door | 2<br>1<br>2<br>71<br>105 | 1 1/2-ft vent<br>1 1/2-ft fuel transfer<br>High-pressure steam<br>Piping<br>Electrical |
| Dresden | 3 | 8 × 8-ft door (equipment)<br>2 1/2 × 6-ft door (personnel)<br>2 1/2-ft-diam door (escape) | 16-ft-diam bolted patch | 145<br>65 | Piping (welded nozzles)<br>Electrical |
| Elk River | 2 | 5 × 7 ft<br>2 1/2 ft diam | 8 × 10-ft bolted and gasketed door | ~250 | |
| Enrico Fermi | 2 | 6 ft 3 in. × 6 ft<br>3 ft diam | | | |
| EBR-II | 2 | 3 × 6 ft<br>5 ft diam | 7 × 9-ft gasketed and bolted door | | |
| EGCR | 2 | 7 ft diam<br>11 ft diam | None | | |
| HWCTR | 2 | 10 ft diam (3-ft 6-in. × 6-ft 8-in. door)<br>3 1/2 ft diam (2 1/2-ft-diam door) | 7 × 7-ft bolted and gasketed door | 2<br>1<br>1<br>2<br>1<br>25<br>180 | 2-ft vent<br>1-ft relief<br>10-in. vacuum breaker<br>10-in. steam<br>16-in. fuel transfer<br>3/4 to 4 in.<br>Electrical |
| HRT | 0 | No access during operation | Two access plugs (30 in.) | 1<br>2<br>4<br>4<br>345<br>272<br>70 | 16-in. vent<br>6-in. dewatering holes<br>1- to 6-in. steam line<br>24-in. porthole<br>1/8- to 6-in. piping<br>Instrument<br>Electrical |

Table 7.5 (Continued)

| Reactor | Air Locks | | Access Openings and Doors | Other Penetrations | |
|---|---|---|---|---|---|
| | Number | Description | | Number | Description |
| Indian Point | 1 | 3-ft 6-in. × 6-ft 8-in. door | Personnel access (7 ft diam)<br>Three 14-ft-diam equipment access openings<br>Six 4-ft-diam emergency exits<br>One 4-ft-diam utility exit | 4<br>2<br>1<br>152<br>316 | 4-ft vent<br>2-ft vent<br>2 1/2-ft fuel transfer opening<br>Piping<br>Electrical |
| Biological shield | | Labyrinth sealed openings with locking doors | | | All above plus annular space ventilation ducts |
| NS Savannah | 1 | 42 in. ID | Four hatches | 2<br>74<br>18 | Vent<br>Piping<br>Electrical |
| Pathfinder | 2 | 3 × 7 ft<br>2 1/2 ft diam | 11-ft-diam bolted and gasketed door | 2<br>1<br>16<br>85 | Vent<br>Fuel transfer<br>Piping<br>Electrical |
| PRTR | 2 | 4 × 8 ft<br>3 ft diam | 5 1/2 × 10 ft | | |
| PWR | 6 | 7 ft diam (2 in each cylindrical tank) | Nine access openings:<br>one, 18 ft diam<br>six, 10 ft diam<br>two, 6 ft diam | | |
| Saxton | 2 | 2 1/2 ft × 6 ft 8 in.<br>2 1/2 ft diam | 6-ft-diam flanged opening | | |
| SM-1 | 0 | No entry during operation | 6 1/2-ft-diam manhole at top, double door access at lower level | | Various piping and conduit |
| VBWR | 2 | 3 1/2 × 5 ft | 10-ft-diam bolted patch | 2<br>3<br>15<br>6 | 2-ft vent<br>10-, 12-, 24-in. steam lines<br>1 1/2- to 12-in. piping<br>Electrical |
| Yankee | 1 | 7 1/2 ft (personnel) | 13-ft 11 1/2-in. gasketed and bolted door | 2<br>1<br>4<br>61<br>213 | 30-in. vents<br>2-ft fuel transfer opening<br>14-in. steam line<br>3/4 to 12 in. (plus spares)<br>Electrical and expansion joints for columns |

Table 7.6. Containment Penetration Closures

| Reactor | Number and Type of Valves Per Line[a] | | | Action of Automatic Valves On Loss of Power | | Parameters Sensed to Close Automatic Valves and Number of Sensors Per Parameter | System Logic For Automatic Valves | Possible Emergency Action if Valve Does Not Close |
|---|---|---|---|---|---|---|---|---|
| | Vacuum Relief | Enclosure Ventilation | Process Lines[b] (Air, Steam, Water, etc.) | Loss of Electric Power | Loss of Instrument Air | | | |
| Big Rock Point | Enclosure ventilation valves open automatically | 2 automatic (normally open) | 1 automatic, 1 manual for lines open to interior of containment; 1 automatic, 1 manual, one for each side of the containment shell for lines open to primary system; lines normally closed have manual control plus lock or interlock | Normally open lines carrying fluids in or out of the containment close, except control rod drive pump supply; ventilation valves close | Normally open lines carrying fluids in or out of the containment close, except control rod drive pump supply; ventilation valves will still operate off accumulator in event of air failure | Ventilation valves, all scram parameters, 4; others: low reactor water level, 4; high enclosure pressure, 4; manual operation possible | 2 out of 4 | Close manually from control room or locally in some cases; close manual backup valves where supplied |
| CVTR | None | 1 manual (normally closed) | 1 manual or 1 local or both on all process lines; 1 automatic and 1 local on radiation monitoring sample inlet and return and instrument air regulator bleed | Radiation monitoring samples (close); instrument air bleed (open) | Instrument air bleed (open) | High enclosure pressure | Any 1 signal to close valves | Close locally |
| Dresden | 2 enclosure ventilation valves open automatically | 2 automatic (normally open) | 1 manual and 1 check in inlet lines; 1 automatic and 1 check in outlet lines | Motor-operated valves switch to station battery; other valves close | Valves close | Enclosure ventilation valves, all scram parameters, 4 or 6; other valves: high enclosure pressure, 4; low reactor water level, 4; manual operation possible | 2-out-of-4 or 2-out-of-6 to close enclosure ventilation valves; 2-out-of-4 for other valves; annunciates only upon high radiation level or steam leak in enclosure; individual lights and annunciator only on enclosure ventilation valves | Close manually from control room and, in some cases, locally also |
| Elk River | 2 | 2 automatic or manual (normally open) | Manual block valves on each line (one valve on each side of containment) | Ventilation valves close (dc) | No action, but loss of service air; ventilation valves close | High stack immediate particulate monitor, 1; high stack delayed particulate monitor, 1; high stack gas, 1; primary system pressure >1210 psig, 1; containment pressure >2 psig, 1; manual operation possible | Any 1 signal to close valves | Can be closed manually from control room |
| Enrico Fermi | 2 | 1 automatic (normally open, except none in closed-circuit nitrogen cooling loops) | 1 manual (locked closed) in purge line; other lines have 1 automatic normally open valve (Note: no water or steam lines in enclosure) | Close | Close | High gas or particulate activity in enclosure or both, 4; high enclosure pressure, 2 | Any 1 of 4 sensors to close all valves; 3 additional sensors close certain selected valves | Close locally |

[a] Automatic indicates a valve closed by instruments. Manual denotes a valve operated remotely by an operator in the control room. Local means a valve operated by hand at or near the valve.

[b] In this tabulation it has not been possible to take into account the pressure rating or special conditions that may apply to the system to which these lines are connected.

Table 7.6. (continued)

| Reactor | Number and Type of Valves Per Line[a] | | | Action of Automatic Valves on Loss of Power | | Parameters Sensed to Close Automatic Valves and Number of Sensors Per Parameter | System Logic For Automatic Valves | Possible Emergency Action if Valve Does Not Close |
|---|---|---|---|---|---|---|---|---|
| | Vacuum Relief | Enclosure Ventilation | Process Lines[b] (Air, Steam, Water, etc.) | Loss of Electric Power | Loss of Instrument Air | | | |
| EBR-II | None | | 1 or more isolation valves per line (automatic) | Close | Close | | | |
| EGCR | 1 | 2 automatic | 2 automatic (normally open) for contaminated water, drain, and sanitary sewer, 1 automatic (normally open) for nonessential services | Close | Close | High stack activity, 3; low reactor coolant pressure, 3 | 2 out of 3 per parameter | Close manually from control room |
| | | | 2 automatic (normally open) from high-pressure helium supply | Open | Open | Low reactor coolant pressure | 2 out of 3 per parameter | Close manually from control room |
| | | | 2 automatic (normally open) from steam, feedwater | | Electric-motor actuated | Low water level in steam generator, 3; high water level in lower plenum of steam generator, 3 | 2 out of 3 per parameter | Close manually from control room |
| HWCTR | 2 | 2 ventilation valves in series in both the supply and exhaust ducts (normally open) | 1 automatic and 1 manual in low-pressure gas vent to exhaust stack; 1 automatic and 1 manual in sump pump discharge line; 2 automatic and 1 manual in high-pressure vent to exhaust stack; 1 automatic three-way butterfly valve in gas-pressure-relief line (normally positioned to relieve to steam discharge lines) | Close, except three-way butterfly valve positioned to relieve gas to inside of containment shell | Close, except three-way butterfly valve; reserve air tank provides for at least 1 operation | High enclosure effluent temperature, 1; high stack activity, 1; manual operation | Any 1 to close valves | Close ventilation valves locally; close additional local valves in process lines where they exist; switch three-way butterfly valve to containment shell with local handwheel |
| HRT | None | None | 1 automatic and 1 manual in main steam lines; 1 manual and 1 check in water inlet lines; 1 automatic in water outlet lines; 1 automatic and 1 manual in cell vacuum line | Varies with valve | Varies with valve | Sump discharge activity; instrument cubicle activity; stack gas activity; steam activity; cooling water activity; cell air activity; oxygen activity; low oxygen pressure activity; high or low fuel system pressure; high sampler pressure; high cell pressure | 1 of 1 for cell vacuum; 2 of 2 for others | Close locally where provision is made |
| Indian Point | 2 | 2 automatic (normally closed) in main system | 2 automatic in lines constituting possible routes for external contamination; 2 manual in vital service lines; 1 automatic and 1 manual in other lines | Enclosure ventilation valves and valves in lines constituting possible routes for external contamination close | Close in lines constituting possible routes for external contamination | High enclosure pressure, 8; manual operation possible; high stack radioactivity for ventilation valves, 1; low pressurizer level for others, 2. | Any one of 4 pairs of sensors close 1 automatic valve of each pair of valves in every nonvital system penetration | Second valve in each penetration provided with electric power from separate sources where electric power is required for closure |
| NS Savannah | None | 1 manual (normally closed) | 1 check valve in inlet lines; 1 automatic in outlet lines; most lines have additional local | Close | Close | High enclosure pressure, 3 | 2 out of 3 to close valves | Close additional local valve where it exists |

Table 7.6. (continued)

| Reactor | Number and Type of Valves Per Line[a] | | | Action of Automatic Valves on Loss of Power | | Parameters Sensed to Close Automatic Valves and Number of Sensors Per Parameter | System Logic For Automatic Valves | Possible Emergency Action if Valve Does Not Close |
|---|---|---|---|---|---|---|---|---|
| | Vacuum Relief | Enclosure Ventilation | Process Lines[b] (Air, Steam, Water, etc.) | Loss of Electric Power | Loss of Instrument Air | | | |
| Pathfinder | 1 and all ventilation valves open automatically | 2 automatic (normally open) | 1 check in inlet lines; 1 automatic in outlet lines | Motor-operated valves switch to station battery; other valves close | Close | Varies with the valves but includes 1 or more of the following: high reactor building pressure, 1; loss of condenser circulating water, 2; high air ejection exhaust radiation, 1; high turbine building ventilation exhaust radiation, 2; high main steam line radiation, 1; high ventilation stack exhaust radiation, 1; high reactor building ventilation exhaust radiation, 1 | Any 1 signal to close valves | Close manually from control room; close additional local valves where one exists |
| PRTR | 2 | 1 automatic (normally open) | 1 check valve in inlet lines; 1 automatic valve in outlet lines | No electrically operated valves in containment | Close | High exhaust air activity, 3; high aqueous effluent activity, 3; manual operation possible | 2 out of 3 | Close manually from control room except for mechanical blockage of valves |
| PWR | | 2 automatic (normally open) | None in hydraulic valve lines; 1 manual in main steam and feed-water lines; 1 manual or local and 1 check valve in other inlet lines | Enclosure ventilation valves close (Note: cannot close enclosure ventilation on loss of hydraulic pressure) | Does not apply | High enclosure pressure, 2; high stack activity, 1; high enclosure air activity, 1; manual operation possible | Any 1 signal to close enclosure ventilation valves only | Close enclosure ventilation valves locally; would take considerable time |
| Saxton[c] | | | | | | | | |
| SM-1[c] | | | | | | | | |
| VBWR | None | 1 automatic; 1 local (normally open) | 1 automatic and 1 local in lines constituting possible routes for external contamination; 1 local in other process lines | All valves close | Enclosure ventilation valves | Varies with valve but includes 2 to 10 of following: high steam radiation, 1; loss of power, 1; high stack radiation, 1; high condenser pressure, 1; low reactor water level, 1; low circulating water pressure, 1; high steam flow, 1; high enclosure pressure, 1; seismic disturbance, 1; high enclosure radiation, 1; manual operation possible | Any 1 signal to close valves | Close local valves, handles outside enclosure (also can be closed manually from control room) |
| Yankee | None | 1 manual (normally closed) | 1 automatic and 1 manual in main steam lines; 2 check valves in other inlet lines; 1 trip valve in other outlet lines | Electrically operated valves close using station battery | Close | High enclosure pressure, 2 | Any 1 to close valves | Close locally |

[c]Information not available.

that allowed the HRT to be a useful research tool. In all, there are more than 700 penetrations.

The piping penetrations listed in Table 7.6 usually consist of a pipe sleeve welded to the vapor container wall through which the pipe line is run and sealed at one or both ends. If this pipe is to be a steam line, which may be subject to thermal expansion, a convoluted (or other type) expansion joint should be provided. Figures 7.29 and 7.30 (ref. 28) show typical examples of both types.

Electrical penetrations are treated similarly in most plants. One of the problems here is to prevent the leakage along individual leads in the one large penetration. Figure 7.31 (ref. 29) shows a penetration containing a number of electrical leads drawn through the normal pipe-type penetration. They are separated from one another by spacers, and the space between leads is filled with some "potting" material. Several "potting" materials are used to prevent the leakage along the individual leads to the outside.

The care with which a penetration is fabricated may be negated by careless management of the line for which the penetration is made. Hanauer has stated:[30]

> "It is clear that containment valves are required to close when they are needed ... . Even a relatively small valve which by failing to close may provide a path directly between the enclosure interior and the atmosphere will leak at a rate very much larger than is specified for the containment. The case for valves in water and steam lines is not so clear. In some cases these lines lead to closed systems whose integrity may exceed that of the containment enclosure. Under normal condition, transport of radioactivity into these closed systems does not constitute a hazard to the general public.
>
> "[Consider the] extreme example ... [of] a once-through air-ventilation system which normally exhausts through a stack. On the occurrence of an accident, valves are to close in the intake and exhaust ducts of the system. If the valve in the intake duct failed to close, any excess pressure in the containment enclosure could be vented directly to the atmosphere through the open intake port. The containment would thus be violated, with activity release perhaps taking place at or near ground level ... . It is worth noting that it might be impossible to close such a valve by direct manual operation because of the high radiation level."

In a containment system there are a large number of pipes and ducts (see Table 7.6), which, upon occurrence of an accident, must be blocked. This great number of penetrations constitutes a danger, since failure to block any one of these could seriously compromise the vapor container. On the other hand, the unintentional closure of valves in piping penetrations could lead to serious damage to the reactor and auxiliary equipment. These remarks are intended to make clear that the systems depended upon to maintain the integrity of the containment should be of extremely high dependability.

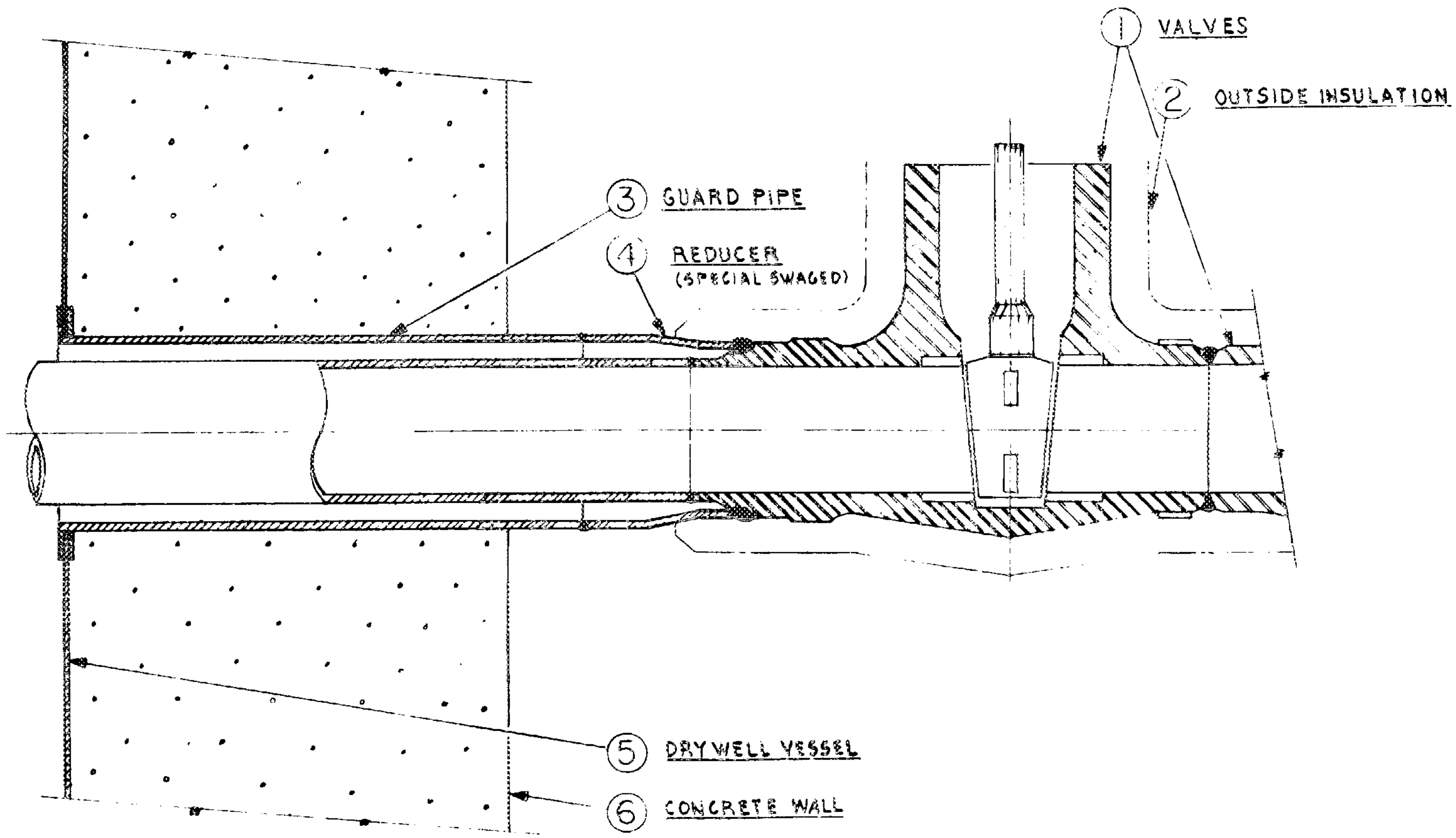



Fig. 7.29. Standard Piping Penetration of the Humboldt Bay Reactor. (From ref. 28)

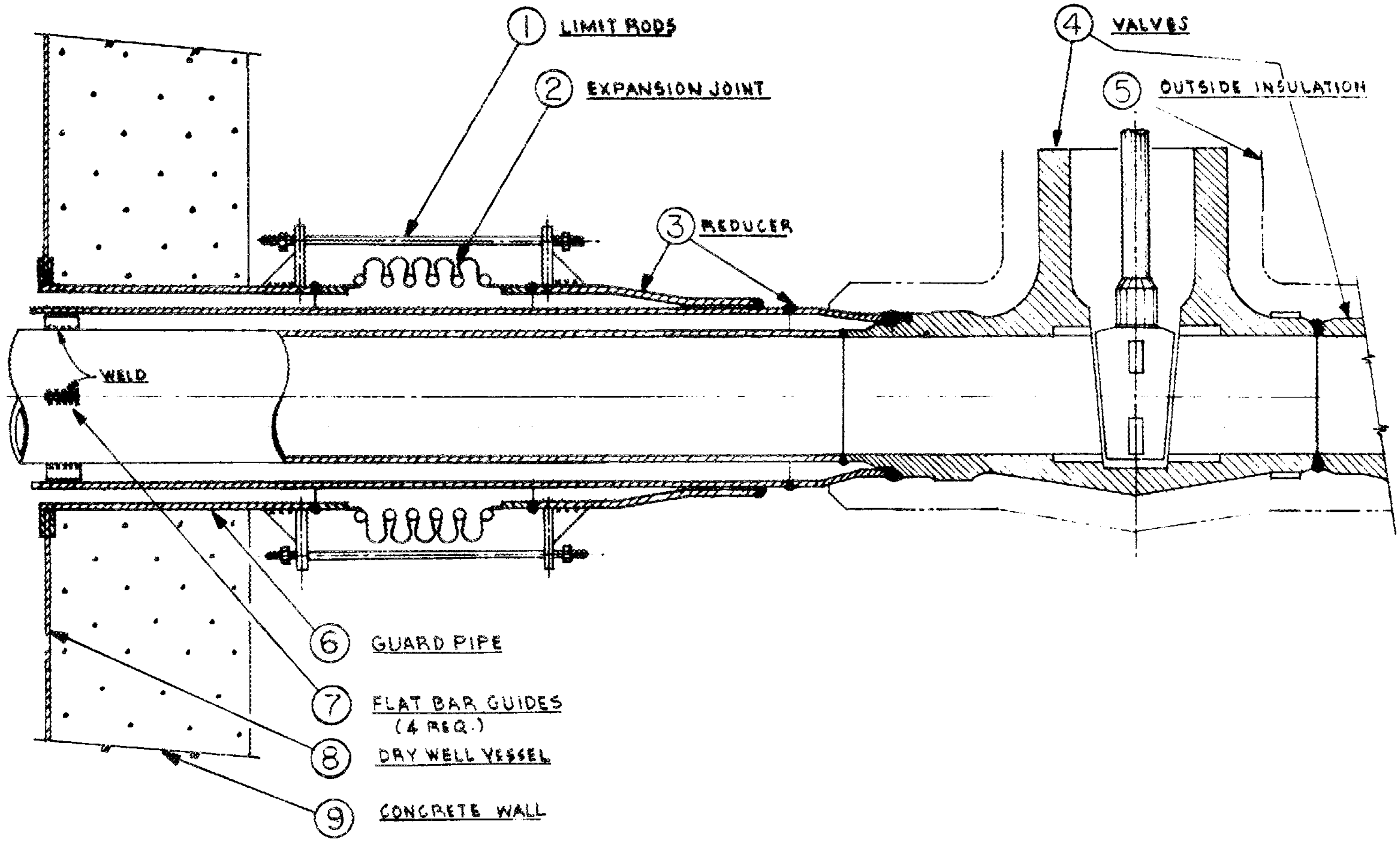



Fig. 7.30. High-Temperature Piping Penetration of the Humboldt Bay Reactor. (From ref. 28)

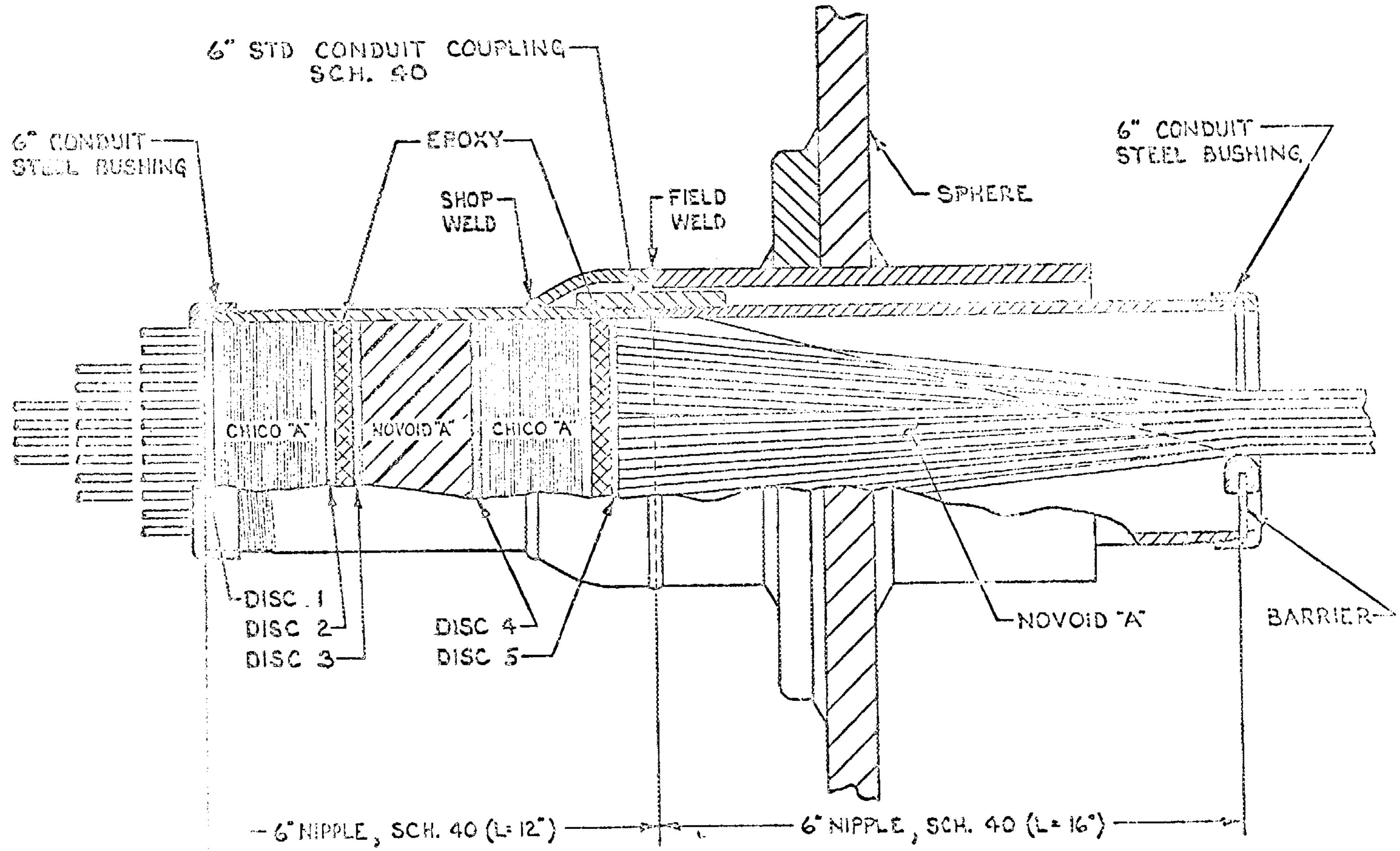



Fig. 7.31. Electrical Penetration Detail. (From ref. 29)

It may be noted from Table 7.6 that a number of parameters are sensed to close various valves. The VBWR is a particularly good example, since it indicates the greatest variety of parameters monitored. It may be further noted that at least two of these parameters must indicate the need for closing the given valve; this redundancy in many cases offers the assurance that an action will not begin if some piece of monitoring equipment is not functioning properly.

The HWCTR ventilation valve is closed by either of two monitors: high temperature in the ventilation duct downstream of the blower or high fission-product activity in the stack. The high temperature (38°F above normal) would indicate in this case a major steam system rupture. The high activity (twice stack background) would obviously show that an activity release of major proportions had taken place. Closure of this valve is afforded by either of these systems, since even accidental closure would not result in any serious consequences.

## 7.2.7 Containment Protection

The container should be protected from the various mechanisms that could render it useless under design conditions. Table 7.7 shows the measures taken to afford protection from weather conditions and internal and external missiles.

### 7.2.7.1 External Protection

In general, some protective covering is provided for reactor containers. This protection may take the form of one of several coats of paint or some of the other commercial coverings such as Amercoat or Bitumastic. In situations where thermal insulation is required, the insulation is installed and then the protective covering is applied to it. Several types of thermal insulation are used (foam glass, polystyrene, etc.). The HWCTR has preshaped polystyrene blocks covered with Fiberglas and epoxy resin and painted with a conventional commercial paint.

The Enrico Fermi containment structure has an elaborate lightning-rod system. The reasoning behind this and the design are as follows:[31]

> "Available lightning data indicate that a stroke of lightning impinging directly upon the Reactor Building would remove some material and could produce a pit approximately 1/16-inch deep by 1/2-inch in diameter. There is no known lightning stroke which could burn through the 5/8-inch thick plate which forms the top of the building. The probability of a direct stroke to the surface has been reduced by the placement of a 16-foot high lightning rod at the top of the building. This provides a 45-degree umbrella to effectively shield the building from a direct stroke to its surface. Ground connections have been provided at 8 points around the periphery of the vessel base. Each of these points is connected to a grounding network which covers the entire area

Table 7.7. Containment Structure Exterior and Missile Protection

| Reactor | Exterior Protection | Missile Protection |
|---|---|---|
| Big Rock Point | Insulation (3/8 in.); cathodic protection | Internal structures provide internal missile protection |
| CVTR | Concrete exterior | Missile protection from the exterior by the concrete building, from the interior by the operating floor and structures below |
| Dresden | Thermal insulation; corrosion protection provided by sprayed insulation covered with paint; cathodic protection | 24 to 60 in. of concrete surrounds the nuclear steam supply system |
| Elk River | Foam glass (2 in.) covered by Bitumastic[a] above grade; sandblasted and painted with Bitumastic below grade | Concrete shielding (2 ft) |
| Enrico Fermi | Asphalt paint below grade; lightning rods | Concrete and steel shields in and below floor; machinery dome above floor |
| EBR-II | Aluminum paint above grade; Bitumastic below grade | By wall and top and bottom closures of the reactor vessel; above main floor, 14 in. of concrete wall and ceiling |
| EGCR | Paint | By design and placement of equipment, no missile will be generated that would penetrate the containment shell |
| HWCTR | Polystyrene blocks covered by fiberglass and epoxy paint | No serious missile generation is feasible simultaneous with primary system rupture |
| HRT | Totally enclosed | Thermal shield above, below, and around the reactor vessel; layers of shield blocks protect in the upward direction |
| Indian Point | Concrete exterior; sphere is coated with Amercoat inside and out (see Fig. 17.9) | 5 1/2 ft concrete around and 1 3/4 ft above reactor; no conceivable missile would possess enough energy to penetrate the shell from the inside because of 3-ft biological shield |
| NS Savannah | Vessel painted and totally enclosed in ship's hold | Highly improbable that a missile of sufficient size to cause damage could be generated; protected from penetration by ship collision by a composite steel and redwood collision mat and concrete secondary shield |
| Pathfinder | Insulated to 5 ft below grade; cathodic protection | Any conceivable missile would be stopped by the structural and shielding concrete (2 ft minimum thickness) |
| PRTR | Below, 1/4 in. membrane; above, 3 in. insulation; cathodic protection | Cylindrical concrete wall 1 ft thick and 33 ft high above the main floor |
| PWR | Totally enclosed | No serious missile generation is feasible simultaneous with rupture of the coolant system |
| Saxton | Paint | Concrete liner (1 1/2 ft) below grade provides missile protection; no additional missile protection is considered necessary |
| SM-1 | Paint on dome, cylinder is concrete | Sufficient missile protection provided by the 2-ft concrete liner |
| VBWR | Paint | Steel (18 in. thick) protects from missiles driven in the upward direction; concrete around vessel protects below the main floor |
| Yankee | Coated with Amercoat | No external missile considered credible; the shell is protected from internal missiles by the concrete biological shielding. |

[a]A coal-tar epoxy paint.

and extends downward to the bedrock below the building foundation. The network is permanently grounded at other points by connections to building piles. This grounding network causes electrical charges to be bled off from the tip of the lightning rod and further reduces the probability of a direct stroke to the building or to appendages."

Many of the buildings are provided with cathodic protection below grade. This system is also known as the "sacrificial anode" system and uses some material that would be more electrochemically active than the containment shell itself. This material is electrically connected to the shell so that any electrolytic action will attack the "sacrificial anode" rather than the containment shell. In addition to the electrolytic protection, the shell is cleaned and coated with a coal-tar type of paint. There are some who believe[32] that the best possible protection is to sandblast, coat with the coal-tar paint, and then cover this with epoxy paint; any further effort would be superfluous.

### 7.2.7.2 Shock Protection

The possibility of the initiation of a shock wave by some means has not been overlooked. In fact much work has been and is being done in this field, as is shown in Chapter 6. However, in the design of reactor buildings (e.g., the vapor container), this has not been considered a significant danger. The transmission of a shock wave in air is quite inefficient, and it is felt that by the time the wave front traversed the several feet (in most cases) of free space between the point of initiation and the container wall, the shock intensity would have decreased to the extent that no harmful effects would be realized. From a different point of view, it is difficult to postulate a "credible" set of circumstances which by their occurrence would initiate a shock wave.

In two cases, however, the possibility of a shock wave was considered, and shock protection was provided. Both are experimental reactors, the HRT and EBR-II. The EBR-II is provided with a blast shield around the sides (2-ft-thick alternating layers of steel and lightweight concrete) and under the bottom (1 ft thick). The top is protected by a 1 1/2-ft gas space (see discussion above). In this case, if some mechanism initiates a shock wave in the reactor itself, the shock (or blast shield) will break up and reduce the force of the shock front before it is able to damage the vapor container.

With the HRT the blast shield was provided as much for protection from missiles as for anything else. It was considered a remote possibility that the pressure vessel (4-in.-thick carbon steel) could suffer brittle failure and by this create missiles and initiate a shock front. From the beginning it was considered that the damaging effects of the missiles that might be created would be much more severe than those caused by the propagation of the shock wave.

More recently it has been the opinion of most experts in the field of metallurgy that brittle failure of a pressure vessel, which has been constructed with the extremely rigid quality control that is applied to nuclear vessels, would not be a credible occurrence. This is, of course, based on the utmost care being taken with the design and manufacture of the vessel. For example, when the pressure vessel of the HWCTR was manufactured, certain of the smaller penetrations were not annealed. This made the designers skeptical as to the NDT temperature of the complete vessel. Samples were taken of the various melts of metal, and various tests were performed. It was the opinion of several reputable metallurgists that the NDT temperature would be no less than 110°F after 20 years of irradiation (see Table 7.4), and this allows a margin of safety.

#### 7.2.7.3 Missile Protection

Missiles may be created by many unusual circumstances. Pipes may break and "whip" around in a container, a thermowell may fail at the weld and be propelled as a missile, or shield blocks could gain enough momentum to become serious missiles. The pressure vessel of the SL-1 was propelled upward by a water-hammer effect with such force that piping connections were severed. The entire vessel moved approximately 9 ft upward before its energy was spent and it fell back into its cavity.

The design of the reactor container must include an analysis of items that could become missiles that could damage the container and then provide protection against them. By sensible placement of equipment, much can be done to preclude the effects of missiles. Missile shields in the form of steel plates or concrete walls of varying thicknesses are sometimes interposed between the potential missile and the vapor container.

Some designers even consider the credibility of external missiles (i.e., an airplane falling on the building or a hit by a missile generated as an act of war). The external force that does seem quite credible is that of a collision with another ship by the NS Savannah. In this case the reactor compartment is protected by special collision shielding (see Table 7.7), as well as the concrete of the radiation shield.

### 7.2.8 Coolant Properties

At least part of the energy that will be released in the event of a maximum accident will be contributed by the reactor coolant. Table 7.8 gives information that pertains to this energy release in the form of mass of coolant and the thermal properties at some set of operating conditions. Other sources that may contribute to the mca are rupture of the secondary system or a chemical or nuclear reaction.

It should be noted in the case of the sodium-cooled reactor that the potential danger does not come from the sensible heat in the fluid, as in the case of water-cooled reactors, but from the propensity of the coolant to react violently with air or water.

### 7.2.9 Energy Sources

This section deals with the energy sources that are made available by some mca and their contribution to the loading of the containment shell.

Table 7.8. Reactor Coolant Properties at Assumed Accident Conditions

| Reactor | Type of Reactor | Moderator | Coolant | Coolant Properties at Assumed Accident Conditions | | |
|---|---|---|---|---|---|---|
| | | | | Quantity (lb) | Temperature (°F) | Pressure (psia) |
| Big Rock Point | Boiling water | $H_2O$ | $H_2O$ | 123,000 | 550 | 1000–1500 |
| CVTR | Pressure tube | $D_2O$ | $D_2O$ | 14,600 | 530 | 1500 |
| Dresden | Boiling water | $H_2O$ | $H_2O$ | 376,000 | 545 | 1000 |
| Elk River | Boiling water | $H_2O$ | $H_2O$ | 28,000 | 540 | 925–1250 |
| Enrico Fermi | Fast breeder | None, graphite reflected | Sodium | 400,000 | 550 (inlet)<br>800 (outlet) | Not pressurized (pump head 78 psig) |
| EBR-II | Fast breeder | None, graphite reflected | Sodium | 640,000 | 700 (inlet)<br>900 (outlet) | Not pressurized |
| EGCR | Gas cooled | Graphite | Helium | 1,350 | 780 | 315 |
| HWCTR | Pressurized water | $D_2O$ | $D_2O$ | 41,000 | 464 | 1200 |
| HRT | Homogeneous solution fuel | $D_2O$ | $D_2O$ | 4,130 | 570 | 2000 |
| Indian Point | Pressurized water | $H_2O$ | $H_2O$ | 163,000 | 500 | 1500 |
| NS Savannah | Pressurized water<br>Primary<br>Secondary | $H_2O$ | $H_2O$<br>$H_2O$ | 66,000<br>8,000 | 508<br>463 | 1750<br>485 |
| Pathfinder | Boiling water with nuclear superheat | $H_2O$ | $H_2O$ | 100,000 | 489 (boiler)<br>825 (superheater) | 600 (reactor)<br>540 (outlet) |
| PRTR | Pressure tube | $D_2O$ | $D_2O$ | | | |
| PWR | Pressurized water | $H_2O$ | $H_2O$ | 131,000 | 500 | 1800 |
| Saxton | Pressurized water | $H_2O$ | $H_2O$ | | 503 | 2000 |
| SM-1 | Pressurized water | $H_2O$ | $H_2O$ | | 595 | 1500 |
| VBWR | Boiling water | $H_2O$ | $H_2O$ | 35,000 | 545 | 1000 |
| Yankee | Pressurized water | $H_2O$ | $H_2O$ | 140,000 | 527 | 2000 |

Table 7.9 briefly describes the accident that is considered to be maximum from the standpoint of energy release. Table 7.10 lists the thermodynamic conditions assumed, quantities of energy released, and mechanisms that might lessen the severity of the accident (i.e., building and core spray systems).

The accidents described in Table 7.9 are those that result in the highest internal pressure. It is this pressure that is used as the design parameter for the containment shell.

In some cases the accident that results in the highest internal pressure will not be the one that releases the greatest amount of fission products to the enclosure. In cases such as this a hypothetical accident or design accident is postulated that takes the high peak pressure of one accident and the activity release from another and compounds them into one accident that results in high pressure and a large release of activity.

Table 7.9. Basis for Containment Design

| | |
|---|---|
| Big Rock Point | Loss of coolant through a double-ended failure of the largest pipe when in hot standby |
| CVTR | Loss of primary coolant followed by 35% Zr-water reaction and $10^6$ Btu nuclear reaction |
| Dresden | Instantaneous loss of all primary coolant |
| Elk River | Instantaneous loss of coolant through largest possible double-ended pipe rupture (10-sec emptying time) |
| Enrico Fermi | Sodium-air reaction; failure of main sodium piping with coincident introduction of air in the normally inert atmosphere; burning continues until 95% of $O_2$ is consumed |
| EBR-II | Ejection of 3000 lb of sodium with a high degree of dispersion |
| EGCR | Loss of primary coolant and the contents of one steam generator |
| HWCTR | Rapid loss of primary coolant through largest credible break (10-in. pipe) |
| HRT | Instantaneous release of core and blanket solutions and the contents of one steam generator |
| Indian Point | Instantaneous release and expansion of all primary fluid and secondary fluid from one steam generator |
| NS Savannah | Rapid loss of primary coolant through break of largest pipe (12 9/16 in. ID) plus the contents of one steam generator; continued operation at 69 Mw for 5 sec |
| Pathfinder | Loss of coolant (assumed instantaneous) during startup at 442°F with superheater flooded |
| PRTR | Loss of primary coolant through rupture in 14-in. top header; complete discharge in 42 sec; metal-water reaction |
| PWR | Loss of primary coolant through a 15-in. pipe break; rupture of the secondary steam generator |
| Saxton | Instantaneous loss of primary coolant plus failure of core spray system (complete meltdown) |
| SM-1 | Loss of primary and secondary coolant at abnormal conditions that allow maximum stored energy release |
| VBWR | Pressure vessel rupture not considered credible, but for calculational purposes the pressure vessel contents assumed to be released instantly; remainder of coolant released in 1.5 sec |
| Yankee | Virtually complete loss of primary coolant through a 20-in. pipe (largest) break; core injection prevents fission-product release (hypothetical accident: same but with fission-product release) |

This is, of course, unrealistic but usually quite conservative. Activity releases will be discussed later.

The accident that initiates the mca is usually the rupture of the coolant system (primary or secondary or both), with subsequent release of coolant. After the rupture, time is required for the contents of the ruptured system to be discharged to the containment atmosphere. During this time, attempering effects on the peak pressure should be experienced, such as heat transfer to cool masses inside the containment vessel, heat transfer to the container shell and to the outside atmosphere, and cooling effects of core or building sprays or of open-water pools. In most cases these effects are not considered when calculating the maximum pressure to be contained by the system.

Table 7.10. Assumptions for Calculation of Containment Structure Design Pressure

| Reactor | Thermodynamic Assumptions | Nuclear Energy Added[a] | Chemical Energy Added[a] | Q, Coolant Stored[a] | Core Spray or Injection | Building Spray | Comments |
|---|---|---|---|---|---|---|---|
| Big Rock Point | Heat loss considered; perfect mixing | None | None | 6.7 | Yes | Yes | Core spray (400 gpm) actuated by low primary system pressure and low reactor water level; building spray (400 gpm) starts 15 min after high sphere pressure is obtained; both sprays can be actuated manually |
| CVTR | No heat loss; perfect mixing | 1 | 35% Zr reaction | 7.31 | Yes | None | Core injection system assumed to fail |
| Dresden | | Enough to melt fuel | 25% Zr reaction | 200 | None | Yes | Building spray automatically actuated after 15 min |
| Elk River | No heat loss; perfect mixing | None | None | 17.1 | Yes | Yes | Operators would stay in control room and take emergency action; control room 480 ft from the reactor; emergency spray starts 10 min after container pressure indicates 2 psig |
| Enrico Fermi | | Negligible compared with chemical energy | 10 | None | None | None | |
| EBR-II | | 0.54 | 18 | None | None | None | |
| EGCR | Heat loss is considered; reactor scrams within 5 sec | None | None | | None[b] | Yes | External building spray cools building shell; actuated when the containment is closed |
| HWCTR | No heat loss; perfect mixing | None | None | 22.1 | None | Yes | Building spray system reduces pressure after mca |
| HRT | No heat loss; perfect mixing | None | None | | None | Yes | Cell spray system must be turned on (administrative action); high cell water level automatically turns it off |
| Indian Point | No heat loss; perfect mixing | 4.3 | None | 130.5 | None | Yes | Automatic containment isolation at 5 psig; at 10 psig, containment external sprays turned on (administrative action); at 15 psig and high gamma, containment internal sprays turned on (administrative action) |
| NS Savannah | Heat loss is considered; perfect mixing | 0.47 | None | 37.4 | None | None | |
| Pathfinder | No heat loss; perfect mixing | None | None | 60 | None | None | Feedwater pumps continue to operate and, in certain cases, mitigate the accident |
| PRTR | No heat loss; perfect mixing | None | 2.6 | | Yes | Yes | |
| PWR | | 4.6 | None | 78 | None | None | |
| Saxton | No heat loss; perfect mixing | None | None | | Yes | None | |
| SM-1 | No heat loss; perfect mixing | 1.2 | None | 7.3 | None | Yes | Manually controlled emergency spray system |
| VBWR | Heat loss for 1.5 sec; perfect mixing | None | None | 20 | None | Yes | Building spray operated manually from outside vessel if needed |
| Yankee | | None | None | 93.3 | Yes | None | Core injection system can supply 5400 gpm of borated water to the core, supplied by a 125,000-gal tank; starts automatically when pressure drops to 1000 psi; assumed to fail for hypothetical accident |

[a]Quantities of heat energy given in millions of Btu.

[b]An emergency core cooling loop is provided, see Sec. 7.9.1.

In calculating the energy balance, the assumptions most usually made are: (1) no loss of heat to structure or equipment, (2) no loss of heat to the shell or transfer to the outside, (3) no loss of energy to a spray or fog system, and (4) perfect mixing of gases and released vapors. Also, the perfect gas law is assumed to apply. It is obvious that since no energy is transferred from the system, it is equivalent to an instantaneous release of all the stored energy of the system. Exceptions to this are shown in the cases of the PWR, EGCR, and the NS Savannah. In these cases it is assumed that heat is transferred from the containment vessel under some set of conservative conditions. In the near future it should become reasonable to take even more credit for some of these attempering effects, since research work is now being carried out in an effort to assign values to these various phenomena.

With respect to water-cooled reactors, the effect of a water hammer must also be considered. For example, the SL-1, an Army low-power boiling-water reactor, experienced a rapid nuclear incident that initiated a water hammer of such force that the vessel was lifted upward about 9 ft. At the time the five boiling-water reactors represented in this discussion were designed or built, the water-hammer effect was not included in the accidents usually analyzed. However, jet forces from broken lines were considered, and as a result the vessels are mounted so as to resist these forces. Whether or not water hammers will be considered in the future remains to be seen.

If it were assumed that the water hammer could occur and, further, could rupture the reactor pressure vessel, then serious revisions would have to be made in the thinking of safety analysts. However, the SL-1 vessel did not fail, and possibly it can be shown that others would withstand such an accident. It appears that two mechanisms present themselves as serious problems. First, there is the possibility of instantaneous release of primary fluid from the ruptured pressure vessel, but in most cases this is assumed anyway. The second is the possibility of missile generation (i.e., blowing off of the top flange) and shock wave generation. This would require special holddown provisions or substantial missile and shock protection.

### 7.2.9.1 Boiling-Water Reactors

The five boiling-water reactors listed in Table 7.9 are examples of reactors in which a major rupture of the largest pipe or vessel and subsequent loss of coolant could occur. There are variations of this accident that depend on the reactor, reactor materials, and the one who is postulating the event. The Elk River, VBWR, and Pathfinder reactor analysts assumed an instantaneous release of all the coolant at the time the most energy was stored in the coolant. This is a conservative approach because any time that elapses during the rise of the internal pressure allows condensation of steam and therefore a reduction in peak pressure.

1. Elk River. For the Elk River reactor, it is assumed that the entire contents of the core and coolant loop are discharged instantaneously to the containment vessel. Although this is an incredible event, it does yield a conservative result. No credit is taken for the attempering effects of the spray system or any heat loss from the shell atmosphere.

Although there is zirconium in the core, it is not used in the fuel cladding, and the reaction of zirconium and water is not considered. Even if it were considered, however, it would add insignificantly to the final pressure.

2. Pathfinder. For the Pathfinder reactor the same assumptions with respect to time of release (instantaneous) and thermodynamic conditions are used. However, this reactor contains a nuclear superheater section, and the time when the maximum amount of stored energy is contained in the reactor is during heatup, just before draining of the superheater section. This occurs when the average temperature of the coolant is 442°F.

3. VBWR. For the VBWR, it is assumed that the primary system fails catastrophically and releases all the reactor vessel contents instantaneously, but the remainder of the loop requires 1.5 sec to empty. During this 1.5 sec, heat transfer takes place to the containment shell so that the peak pressure is suppressed by about 5.5% of what it would have been if all the coolant had been released instantaneously.

4. Big Rock Point. For the Big Rock Point reactor, it is assumed that the largest line to the reactor ruptures and releases the contents of the primary system in the shortest possible time. The containment atmosphere conditions assumed at the time of the accident are a temperature of 100°F and 100% relative humidity. Heat transfer is assumed to take place to the cold masses of the containment structure until an equilibrium temperature is attained. Heat transfer coefficients of 700 and 240 Btu/hr·ft$^2$·°F for steel and concrete, respectively, are assumed.[8] The peak pressure of about 20 psig occurs approximately 16 sec after the pipe breaks.

5. Dresden. The mca assumed for the Dresden reactor involves the instantaneous and complete severence of one of the bottom inlet lines while the reactor is in the hot standby condition and subsequent release of all primary coolant. Approximately 1 min after the accident the peak pressure would be reached.[33]

### 7.2.9.2 Pressurized-Water Reactors

With pressurized-water reactors, as with boiling-water reactors, the maximum accident is the rapid release of water from the cooling systems. A complete loss of primary coolant, with an accompanying loss of part or all of the secondary coolant, through the largest credible rupture is postulated for most of the reactors. In some cases the release is taken to be instantaneous, since it is felt that this is even more conservative than discharge through the largest credible rupture. Yankee, Saxton, and HWCTR consider only release of the primary fluid in the maximum accident (i.e., no failure of the primary heat exchanger).

In none of the pressurized-water reactors is any addition of heat assumed because of a metal-water reaction that would add to the initial pressure peak. In the HWCTR it is assumed that all the uranium and zirconium in the fuel elements will undergo a steam-water reaction, but this is considered to take place at a slow rate and not to contribute to the peak pressure. For the NS Savannah the reactor is assumed to remain at power for a short time after the initial pipe break, thus adding a small amount of energy from the nuclear source. This is not, however, a nuclear excursion.

To be as conservative as possible, for five of the reactors (SM-1, Indian Point, Saxton, HWCTR, and HRT) it is assumed that the released steam and water exhibit perfect mixing with the cell atmosphere and that no heat is transferred to the containment shell or to equipment or structures inside the shell. Further, in each case the mitigating effect of the building or core spray system is not considered. None of the reactors take credit for spray systems in the reduction of the pressure peak, but some do take into account the fact that heat would be lost to the building shell and to equipment in the building.

1. SM-1. A series of events before rupture of the primary system that would allow the reactor system to obtain the maximum amount of stored energy was postulated for the SM-1 reactor. By misoperation and failure of several safety devices the SM-1 would acquire an amount of stored energy much in excess of that contained under any normal operating condition. For other reactors, the normal operating conditions at which the highest energy content is available are assumed; that is, the normal conditions under which the average coolant temperature is the greatest.

2. HWCTR. No credit is taken for the reduction of the pressure peak by the building spray in the HWCTR, but it is postulated that the building spray operation would reduce the containment pressure to atmospheric in 12 hr. From the standpoint of fission-product release to the environs, the accident lasts only 12 hr.

3. NS Savannah. The NS Savannah reactor is unique among the existing reactors in that the pressure vs time history of the containment atmosphere is studied closely. Figure 7.32 (ref. 34) shows this relation.

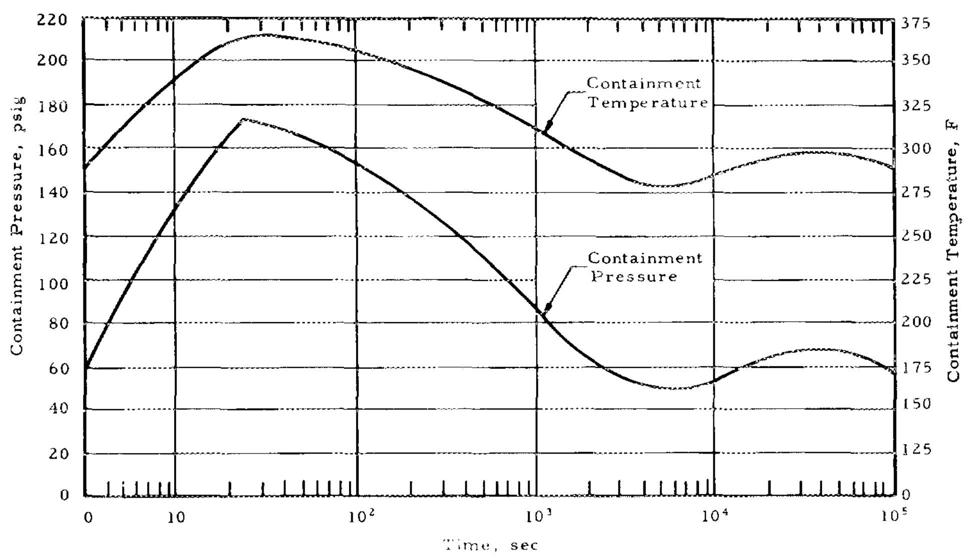



Fig. 7.32. NS Savannah Containment Temperature and Pressure After Primary System Rupture. (From ref. 34)

The sources of the energy added to the containment are (1) the energy contained in the water of the primary system and in the water and steam of one steam generator (secondary system) and (2) the energy of continued full-power [69-Mw(t)] operation for a period of 5 sec after the initiation of the accident. The primary and secondary fluids are to issue from the largest credible rupture in the shortest reasonable time. The cell air-conditioning system and the steam generators cease immediately to remove heat from the system; the only available cooling is from condensation of steam on the exposed surfaces inside the vapor container. Although this is not the "no heat loss" (adiabatic) analysis, the above assumptions do maximize the pressure buildup from the point of view of the mechanistic type of analysis.

### 7.2.9.3 Pressure-Tube Reactors

The pressure-tube reactor is similar to the pressurized-water reactor, with the major difference being the amount of water pressurized and circulated. In the pressurized-water reactor the coolant and the moderator are the same fluid, and this is circulated as the primary coolant. The pressure-tube reactor consists of fuel bundles contained in several small tubes (3 to 4 in. ID). These tubes are immersed in a tank of water called the moderator tank. Pressurized water is circulated through the tubes to remove heat from the fuel, and the large bulk of water that is not pressurized in the moderator tank serves to moderate the neutrons. Thus the inventory of circulating pressurized water is greatly reduced, and if the tubes are made of a low-cross-section material, such as an alloy of zirconium, there is no great loss of neutrons. The PRTR uses Zircaloy-2 for its pressure tubes, and the CVTR uses Zircaloy-4.

Some credible rupture through which the primary coolant must discharge is postulated for both these reactors, but the calculation of the peak pressure is made assuming instantaneous release of all the coolant plus any additional contributions. For both the reactors a large contribution of energy by the chemical reaction of water with the fuel cladding material is assumed. A small nuclear excursion is also assumed for the CVTR. In both cases, assumption of perfect mixing, no heat transfer, and no attempering effects of the spare systems were made.

### 7.2.9.4 Sodium-Cooled Reactors

The design accident for the EBR-II is the discharge, with good dispersion of the molten sodium, of a part of the metal coolant. The primary coolant system contains 640,000 lb of sodium, and the part released constitutes 0.5% of that amount. The postulated accident was based on the work done by the Argonne National Laboratory regarding sodium-air reactions; it represents reasonable and realistic assumptions.

### 7.2.9.5 Gas-Cooled Reactors

The most serious accident to be experienced with the EGCR, as with most other gas-cooled reactors, is the loss-of-coolant accident. This accident, often called the rapid depressurization accident, consists of a primary coolant system rupture that allows the loss of cooling gas and a loss of system pressure.

Since loss of coolant does not constitute a loss of moderator, the nuclear reaction will continue until the control rods are inserted. It is assumed that the loss-of-coolant accident will in no way damage the safety system, and the reactor will be scrammed by the accident. Even with the reactor shut down, temperature transients caused by decay heat will cause fuel elements to fail and expel the contained fission products. Calculations were made to determine the maximum number of EGCR fuel elements that would rupture and discharge fission products.[35] For the EGCR it is also considered that a major rupture in one of the two steam generators might occur that would release its contents to the containment structure and thus contribute to the pressure rise.

The building spray system of the EGCR is similar to that used on the Indian Point Reactor. Instead of spray nozzles inside the building that would produce a fog, which would not only reduce the pressure but also wash out some fission products, the EGCR has only sprays that would douse the outside of the dome of the building. The cooling of this spherical dome would provide heat removal by increased heat transfer through the steel chamber.

### 7.2.10 Fission-Product Release and Dispersion

The effects of the release of fission products in an accident of major proportions are discussed below. Table 7.11 lists the fission products that would escape to the container and the effects of direct radiation from this source. Table 7.12 lists those released from the containment building and the submersion and inhalation doses that may be expected. Table 7.13 gives the rate of building leakage and the meteorological conditions that will cause the transport of the released fission products. The information presented in these tables was obtained from the hazards summary reports listed in the bibliography (Sec. 7.10) unless otherwise specified.

#### 7.2.10.1 Power History

The amount of fission products contained in the core at a given time is dependent upon the power history of the reactor prior to that time. The most conservative estimate would be infinite operation at full power, which would yield the greatest burnup and therefore the highest fission-product inventory. In all cases listed in Table 7.11 this assumption is closely adhered to, although in the case of the NS Savannah the power history may be varied and take into account the particular port that it is to visit (i.e., if a given port can supply only a small exclusion distance, the power of the ship's reactor may be altered so that upon reaching port the fission-product inventory will be small compared with that after full-power operation).

Table 7.11. Fission-Product Release in the Event of an Accident

| Reactor | Operation Assumed Before Accident | Released to Container (%) | Released from the Building (%) | Comments |
|---|---|---|---|---|
| Big Rock Point | 240 Mw(t), long term | 100 noble gases<br>100 halogens<br>50 volatile solids | 100 noble gases<br>50 halogens<br>15 volatile solids | Release to be in two steps: first 20% of the gases when cladding perforation temperature is reached, remainder when fuel meltdown temperature is achieved; assume 10% core meltdown; fractional removal rates of $1 \times 10^{-3}$/sec for halogens and $3 \times 10^{-4}$/sec for solids are assumed |
| CVTR | 65 Mw(t), 3 months | 100 noble gases<br>50 halogens<br>1 strontium | 100 noble gases<br>50 halogens<br>1 strontium | Instant dispersion of fission products in container; 100% of volatiles considered for direct dose |
| Dresden | 626 Mw(t), long term | 100 noble gases<br>100 halogens<br>30 solids | 100 noble gases<br>(see comments for halogens and solids) | A fractional removal rate of $9 \times 10^{-5}$/sec for halogens and $3 \times 10^{-4}$/sec for solids is assumed in the calculation of this release |
| Elk River | | 8.62 noble gases<br>8.62 halogens<br>0.43 strontium | 8.62 noble gases<br>8.62 halogens<br>0.43 strontium | Emergency cooling limits melting to 8.62% of the core; cladding begins melting at 11.67 min; at 10 min, pressure is down to 11 psig |
| Enrico Fermi | 200 Mw(t), long term | 100 noble gases<br>50 halogens<br>5 solids | 100 noble gases<br>25 halogens<br>5 solids | |
| EBR-II | 60 Mw(t), 135 days | 1 of all fission products | 25 per day of all fission products | |
| EGCR | 83.4 Mw(t), 520 days | 0.02 noble gases<br>0.013 halogens<br>0.008 solids | 0.02 noble gases<br>0.013 halogens<br>0.008 solids | Release of contents of 340 failed fuel elements; release of 0.04% of total activity |
| HWCTR | 70 Mw(t), 365 days | 100 noble gases<br>50 halogens<br>1 strontium<br>1 solids | 100 noble gases<br>50 halogens<br>1 strontium<br>1 solids | |
| HRT[a] | | | | |
| Indian Point | 585 Mw(t), 470 days | 100 noble gases<br>100 halogens<br>10 of the rest | 100 noble gases<br>100 halogens<br>10 of the rest | Annular space is swept out and discharged up the 143-m stack |
| NS Savannah | Dependent on port | 100 noble gases<br>50 halogens<br>1 solids | 100 noble gases<br>25 halogens<br>0.0005 solids | Reactor compartment kept at negative pressure; swept out gases directed through filters and up stack |
| Pathfinder | Long-term operation | 75 noble gases<br>25 halogens<br>1 solids | 75 noble gases<br>25 halogens<br>1 solids | |
| PRTR | Pu: 10,000 Mwd/adjacent ton | 100 noble gases<br>25 halogens<br>15 volatile solids<br>0.3 solids | 100 noble gases<br>25 halogens<br>15 volatile solids<br>0.3 solids | |
| PWR | 270 Mw(t), 3000 hr | 2.4 noble gases and halogens<br>0.3492 solids | 2.4 noble gases and halogens<br>0.3492 solids | |
| Saxton | 28 Mw(t), <200 hr | 50 noble gases<br>25 iodine<br>0.5 strontium and cesium | 50 noble gases<br>25 iodine<br>0.5 strontium and cesium | Safety injection system fails; spray fails to work |
| SM-1[a] | | | | |
| VBWR | Extended time at 50 Mw(t); equilibrium concentration of fission products | 100 noble gases<br>100 halogens<br>30 solids | 100 noble gases<br>100 halogens<br>30 solids | |
| Yankee | 600 Mw(t), infinite | 20 gases and volatiles | 20 gases and volatiles | Instantly released to the vapor container |

[a]The assumed maximum accident release is a catastrophy and does not apply well in this work.

Table 7.12. Doses from Fission-Product Release from Containment Structure

| Reactor | Direct Radiation Dose[a] | From Cloud[a] | Thyroid[a] | Bone[a] | Comments |
|---|---|---|---|---|---|
| Big Rock Point | 0.06 rad | 0.06 rad | 4 rem | <0.1 rem | At 0.5 mile for entire accident; postincident spray works |
| CVTR | 0.1 mrem, 2 hr | 6.5 rems, 2 hr<br>0.4 r, 1 day, 1/2 mile | 29.4 rems, 2 hr<br>46 rems, 2 hr | | The 29.4-rem thyroid dose is for a train stalled 300 ft from vapor container (see discussion) |
| Dresden | 2 rems in first 24 hr | 10 rems in first 24 hr | 2400 rems in first 4 hr | 40 rems in first 4 hr | Distance, 1/2 mile |
| Elk River | 0.03 r first hr, 1/4 mile (skyshine) | 0.2 r, 8 hr | 28 r, 1 hr | 0.4 r, strontium, first hour | Building spray is depended upon to reduce the building pressure |
| Enrico Fermi | 0.051 rem, 2 hr<br>5.6 rems, 1 yr | 0.94 rem, 2 hr<br>2.0 rems, 1 yr | 88 rems, 2 hr | 5.7 rems, 1 yr | |
| EBR-II | ~11 r, 1 hr at building surface | 3.19 r, 2 hr | 69 rads, 2 hr | | At a distance of 100 m |
| EGCR | $2.4 \times 10^{-6}$ r, 2 hr | 0.004 rem, 2 hr | 0.33 rem, 2 hr | 0.023 rem | |
| HWCTR | 0.1 r, 10 hr, 3300 ft | 25 rems, 12 hr, 3 miles | 300 rems, 12 hr, 3 miles | | When the iodine removers are put into operation, this will be considered to mitigate the accident release[b] (also see discussion) |
| HRT[c] | | | | | |
| Indian Point | ~0.02 mr/hr, 1 hr, first week | ~3400 mr gamma, 7000 mr beta, 1 hr | 32 r, 1 week | 4 r | |
| NS Savannah | 1.8 rems/hr, A-deck aft of reactor compartment | <25 rems | <300 rems | Negligible | Doses are at controlled-zone boundary; zone radius and exposure duration depend upon particular port |
| Pathfinder | <0.1 r, 1 hr, 1/2 mile | 0.1 rem, 1 hr | 2.2 rems, 1 hr | 2 rems, 1 hr | 1/2 mile |
| PRTR | 48 r, 10 hr, 660 ft | 15 mr/hr, 1 day, 660 ft | | | Limited access to a path 13,000 × 550 ft downwind |
| PWR | Container underground | 10 r beta, 0.9 r gamma, 12 hr, 1/4 mile | 5000 reps I, 167 weeks, 1700 ft | 0.5 reps, $Sr^{90}$ | No credit from iodine plate-out; at an equilibrium dose rate of 0.3 rep/week, the period over which the dose must be integrated to obtain 50 reps is then 167 weeks |
| Saxton | 9 r, 8 hr | 0.8 rad, 8 hr | 208 r, 1 hr | 84 mr | |
| SM-1[c] | | | | | |
| VBWR | 7 r in 3 hr, 600 m | 4 r, 3 hr | 100 r, 3 hr | | Thyroid calculation – iodine decay is considered; distance, 600 m |
| Yankee | <6 r, 1 hr, 1400 ft | | 36 rems, 8 hr, 4000 ft | | |

[a]At exclusion fence unless otherwise stated.

[b]The HWCTR is unable to keep the rate of building leakage below the required minimum, so these iodine removers are to be installed. It is believed that the leakage difficulties are due to the type of construction, which is a departure from the normal approach. It is felt that these problems can be dealt with in time, however.

[c]The assumed maximum accidental release is a catastrophy and does not apply well in this work.

Table 7.13. Dispersion Conditions Assumed for Accident Analyses

| Reactor | Assumed Leakage Rate | Height of Release (m) | Meteorological Conditions | Wind Speed (m/sec) | Site Boundary Distance | Dispersion Parameters | |
|---|---|---|---|---|---|---|---|
| Big Rock Point | Function of sphere pressure based on 0.5%/day at 27 psig | 0[a] | Inversion | 1 | 1/2 mile | Not applicable | |
| | | | Inversion | 5 | | Not applicable | |
| | | | Neutral | 5 | | Not applicable | |
| | | | Stable | 5 | | Not applicable | |
| CVTR | 0.1%/day | 0 | Stable | 3 | 1/2 mile | n = 0.55 | $C_z$ = 0.05<br>$C_y$ = 0.4 |
| Dresden | A function of sphere pressure | 10 | Average | 5 | 1/2 mile | n = 0.25 | C = 0.2<br>$C^2$ = 0.014[b] |
| | | | Inversion | 1 | | n = 0.5 | C = 0.05<br>$C^2$ = 0.004 |
| Elk River | 0.1%/day at 21 psig | 0 | Inversion | 1 | | n = 0.5 | $C^2$ = 0.002<br>$C_z$ = 0.02 |
| Enrico Fermi | 0.15%/day for first 2 hr; 0.10%/day for remainder of one year | | Inversion | 1 | | n = 0.55 | $C_z$ = 0.08 |
| EBR-II | | | | | | n = 0.55 | $C_z$ = 0.08 |
| EGCR | 0.3%/day for 24 hr at 9 psig; 0.2%/day at 4.2 psig for an additional 37 hr; then zero | | Lapse | 2.3 | | n = 0.23 | $C_z$ = 0.3 |
| HRT | See discussion | | | | | | |
| HWCTR | 1/2%/day for 12-hr period, at which time the pressure is assumed to have reduced to the point that leakage no longer occurs | 0 | Inversion | 0.45 | 3 miles | n = 0.5 | $C_z$ = 0.05 |
| Indian Point | 0.1%/day; leakage rate decreases after the first day due to pressure reduction | | Inversion | Low | 0.3 miles | n = 0.2 | $C_z$ = 0.5 |
| NS Savannah | 1.5%/day for 24 hr; it is considered that the ship can be moved in event of accident | 0 | Inversion | | | Completely dependent upon the particular port | |
| PRTR | | | Strong inversion | 1 | | | |
| PWR | Leakage rate is a function of pressure; vol %/min = $1.395 \times 10^{-5}(P)^{1/2}$ | 0 | Inversion | 1.3 | | n = 0.5 | $C_z$ = 0.05 |
| Saxton | 0.2%/day; this rate continues for 8 hr, at which time the pressure is assumed to decrease | 0 | Inversion | 3 | | n = 0.55 | $C_z$ = 0.05 |
| SM-1 | See discussion | | | | | | |
| VBWR | 1%/day except for inhalation dose when reduction of pressure is taken into account | 0 | Average inversion | 5 | | n = 0.25<br>n = 0.5 | C = 0.12<br>C = 0.06 |
| Yankee | 70 cu ft/hr assumed indefinitely | | Inversion | 1.2 | | n = 0.25 | $C_z$ = 0.20<br>$C_y$ = 0.25 |

[a]Effect of building wake taken into account.

[b]This analysis was considered obsolete by General Electric.

The VBWR hazards report specifies equilibrium concentrations of fission products, and the Big Rock Point report specifies long-term operation at full power. The others specify a long operating time at full power such that the maximum expected fission-product inventory will be present at the time of the postulated accident.

### 7.2.10.2 Fission-Product Release

As may be seen in Table 7.11, the assumed release of fission products varies from one installation to another. The fission-products released, as set forth in the example in Appendix C, Part 100, Title 10, of the Code of Federal Regulations, are 100% of the noble gases, 50% of the halogens, and 1% of the solids. This is the release assumed for a pressurized-water type of reactor. Credit is given for the removal of halogens by plating out on the container walls, which reduces the free halogens to 25% of the total. The leakage calculations are based on this inventory of fission products in the free volume. The direct dose, however, is (or should be) calculated using all the fission products that were released from the core.

Notable exceptions to the above postulates of fission-product release from the fuel are those for the PWR, Elk River, Humboldt Bay, and EGCR, for which the problem was approached from the mechanistic point of view. In this type of analysis, tedious calculations are made concerning the temperature and pressure transients in the core and individual fuel elements that lead to the rupture of the fuel element cladding.[35] From such a calculation, it is found that some portion (not total core melting) of the fuel elements will fail, and all the noble gases and halogens (50% halogen release from EGCR) and a small portion (1 to 5%) of the solids will be released from the failed fuel element. This is obviously less conservative and probably more realistic than the nonmechanistic approach used in the example in the Code of Federal Regulations. For other reactors, such as SM-1, Yankee, and Saxton, it is judged that something less than 100% of the fission-product inventory will be released to the container. The estimates for the release to the container for the Pathfinder reflect the fact that some of the fission products will be trapped to form compounds with water and water vapor (washdown) within the shell and others, even though dispersed in the container, will be of such particle size that leakage through the available pores will be impossible.

In Tables 7.1 through 7.13, the HRT and the SM-1 stand alone because the assumed accident is significantly different. These are experimental reactors located in sparsely populated areas, and the exclusion area is very large. In view of these facts the accident considered consisted of a catastrophe that ruptured the container so that a large percentage of the total fission-product inventory was released instantaneously as a cloud. As far as the users of this document are concerned, this is not a credible or even reasonable assumption to be included in this treatment of hazards.

### 7.2.10.3 Direct Radiation

The gamma dose resulting from radiation directly from the fission-product source within the container is available for most reactors. Generally, the assumption is made that all the fission products that are released are dispersed uniformly throughout the building and the portion that is above ground (the earth is considered to act as a shield for the below-ground portion of the building) is taken as the source. Concrete structures that are part of, or adjacent to, the containment shell are considered as shielding; the steel shell (usually thin, 3/8 to 1 1/2 in.) is not generally considered in the shielding for this calculation. Since the PWR is below ground, no direct radiation calculation was made.

One of the most peculiar containment structures with regard to shielding is at Indian Point. This is a conventional steel sphere that is housed inside a concrete building with 5 1/2-ft-thick walls and an arched roof about 2 3/4 ft thick primarily for reduction of the direct radiation dose. This type of shielding also has an advantage with regard to leakage from the shell. This will be discussed in a later paragraph.

For an accumulated or integrated dose at a given point, it may be of significance to take into account the decay of the contained isotopes. Quite obviously, as the fission products decay, the source strength and, consequently, the dose will be reduced. In some cases this was considered.

### 7.2.10.4 Leakage from the Building

The release of activity from the container after the fission products have been released from the fuel and distributed throughout the containment atmosphere is also studied. It has been found that much of the harmful activity will be sorbed at various points inside the container. This is discussed more thoroughly in Chapter 4. The example shown in the Code of Federal Regulations allows for removal of half the halogens without giving consideration to washdown that will occur from the operation of the fog system or the effects of filtering the container atmosphere. This sort of analysis permits specifying the maximum concentration of fission products available for leakage from the container; the rate at which the container leaks is taken up in the next section.

The leakage from the containment structures of the NS Savannah and Indian Point can to some degree be controlled (see Sec. 7.9). Since there is a building around the containment shell in each of these cases, the containment shell would leak fission products into the building and then into the atmosphere. Because this additional holdup is provided, the concentration of fission products leaked to the environs would be lessened.

Difficulty has been experienced at the HWCTR in maintaining the specified container leakage rate; as a result of this, efforts are being made to reduce the seriousness of a release. It is proposed that four iodine cleanup loops be placed in the containment system for the purpose of lessening, to the greatest degree possible, the amount of harmful fission products released from the containment. This proposed system is to be of high reliability. The four units are to be completely separate so that there is no interdependence for effective operation.

Many assume that the fission-product inventory available for release to the atmosphere is the same as that released from the fuel element, whereas others take credit for the fact that some of the fission products may be removed by washdown or plateout. Generally speaking, credit may be taken for decay of fission products while within the containment vessel and during the time necessary for the radioactive cloud to reach the recipient. In some cases, no decay at all of the fission products is considered.

### 7.2.10.5 Leakage and Transport

Some value or system of values for the rate at which the building is assumed to leak, together with an assumed "worst" set of meteorological conditions, determines the doses to be expected by an off-site receiver. In general, upon the occurrence of the mca, it is assumed that the containment structure leaks its contents at some specified rate for some specific length of time. For example, in the EGCR, a leakage of 0.3% per day is assumed for the first 24 hr and 0.2% per day for an additional 37 hr. After this the leakage is considered to be zero. As may be seen in Table 7.13, assumptions for the Enrico Fermi Reactor are similar, except that the lower leakage rate is assumed to continue for a full year. Another example of the limited leakage rate is the NS Savannah. This container is assumed to leak 1.5% per day; but it is also assumed that, owing to the mobility of the ship, it may be towed out to sea or away from the general public. For most of the reactors the leakage is initially some specified value at some internal containment pressure. Then the pressure may be expected to decrease, and so the leakage rate would also decrease. This pressure vs leakage relation is usually taken to be conservative, that is, the greatest leakage for a given pressure (see also Chap. 10).

The leakage is assumed to be a representative sample of the container atmosphere, and since the concentration of activity in the atmosphere is specified by the assumed release (Table 7.11, column 4) and the container free volume (Table 7.2), the activity that escapes from the container is known. This activity will disperse under the influence of the meteorological conditions that prevail. The dispersion of the escaped gases and particles is assumed in most cases to follow the relation developed by Sutton.[36] More recently, however, the relationships developed by Gifford[37,38] have proved to yield more valid results (see Chap. 4 for a more detailed discussion of atmospheric dispersion).

The stability parameter specified by the weather conditions will influence the ground concentration at some distance from the reactor. For example, the EGCR specifies two "worst" weather conditions for the mca (see Fig. 7.33, ref. 39); persons on public land (distance of ~1000 m) near the site boundary would get a high dose under lapse conditions and nearly no dose under inversion conditions. Those at a distance of 10,000 m would get little dose under lapse conditions and a high dose under inversion conditions. The reasons for this may be clearly seen in Fig. 7.34 (from ref. 40). Under lapse conditions the stack effluent is dispersed quickly to the ground, whereas the inversion allows more gradual dispersion. Only for Indian Point and the EGCR is it assumed that the activity

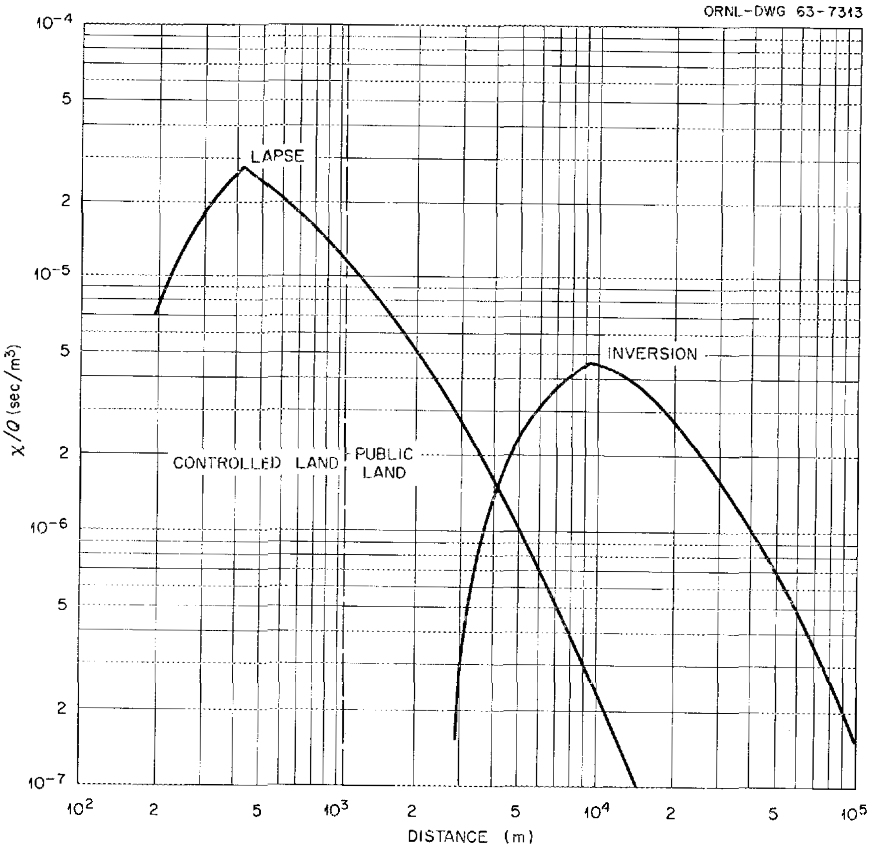



Fig. 7.33. Dose Factors as a Function of Distance from the EGCR. (From ref. 39)

is released up a high stack; a ground-level release (or near ground level, see Dresden, Table 7.13) is specified for the others. This release, under inversion conditions, would yield the most severe doses (i.e., the least dispersion of the activity).

For the Big Rock Point reactor, an interesting departure is made from the usual assumption of the Sutton equations. The analysis is based on inversion conditions under which the greater the distance from the source the greater the dispersion, and so the smaller the dose. In this case it is assumed that the building leaks. The leaked material moves into the wake of the building and is immediately dispersed by the turbulence induced by the presence of the building. This dispersion can be simulated by assuming that the source has been moved away so that this point may be assigned some downwind distance, x. This may be called a virtual source distance, and it serves to decrease the calculated downwind exposures for a given distance from the real source.

Calculations of submersion dose in the cloud and direct radiation dose from the cloud are made using the aforementioned Sutton formula and those developed by Gifford for the computation of the sources. Very adverse weather conditions with regard to dispersion are considered for these calculations.

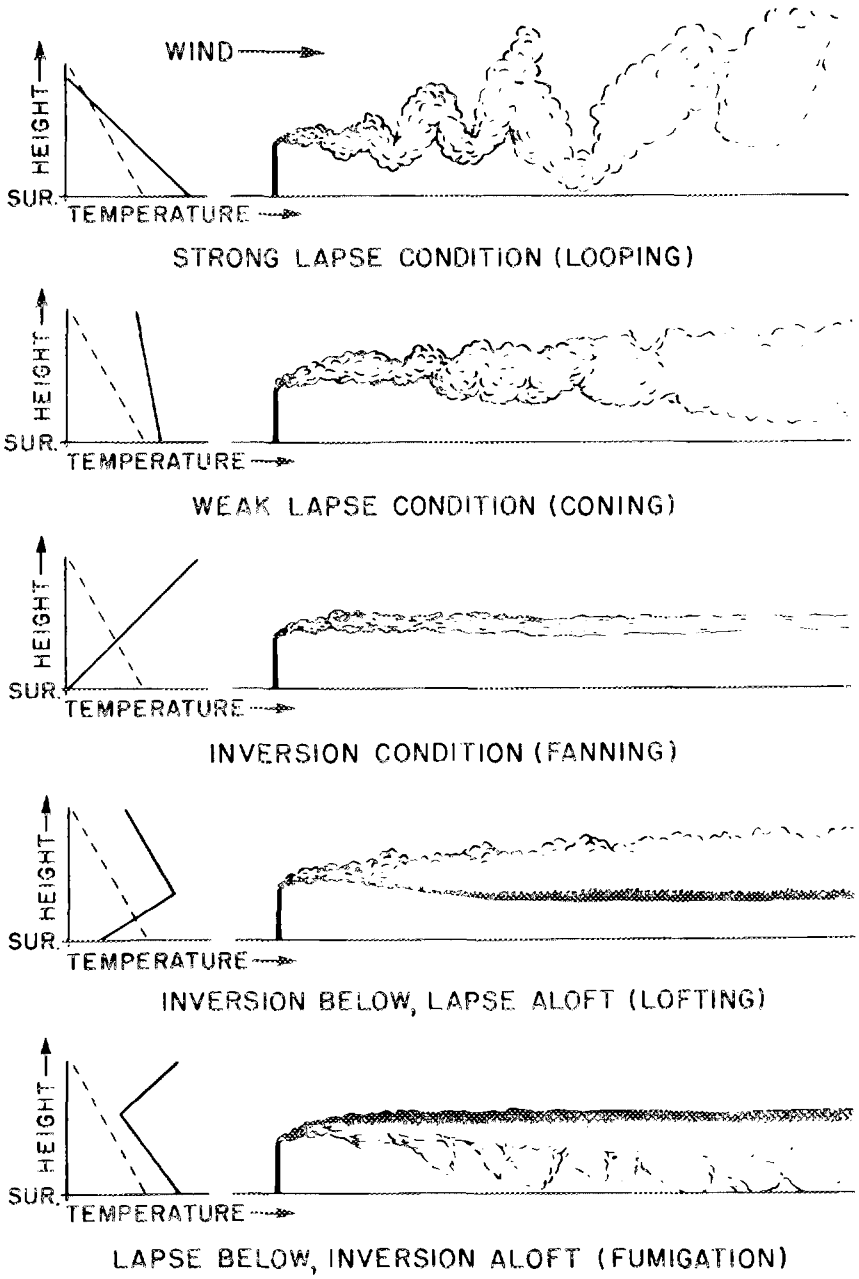



Fig. 7.34. Stack Effluent Behavior at Various Weather Conditions. (From ref. 40)

A peculiar occurrence was considered in the analysis for the CVTR. A railroad track is located about 300 ft from the container. It was assumed that a train could stall at this point; and, although no direct radiation would be suffered because of the hill that is interposed between the reactor and the track, under rare conditions the submersion dose might be large. Meteorological conditions that might exist in the early morning would give rise to dispersion of fission products (if released) in a manner known as "mechanical fumigation." This would allow localized dispersion to occur at the site of the stalled train and would deliver the doses shown in Table 7.12. This has been found to follow the relation[41]

$$\chi = \frac{Q}{\pi C_y \bar{u} H x^{(2-n)/2}} ,$$

where H is thickness of the cloud layer. The other terms were defined previously (see Chap. 4).

## 7.3 LOW-PRESSURE CONTAINMENT

There is great similarity between high-pressure and low-pressure containment. In both the purpose is to contain all the energy and fission products that may be released during the maximum accident. The difference is in the magnitude of the pressure to be contained. It is obvious that for the lower pressure, the costs and leakage difficulties associated with the vapor container decrease. A further advantage is that the low-pressure container offers less hazards when pressurized and is accordingly governed by a less-strenuous code. However, the low-pressure container has not been widely used because most reactors require high-pressure capability if a container of reasonable size is employed; therefore only two low-pressure examples (BONUS and Piqua) are presented in Tables 7.14 through 7.26. Of these two, only the BONUS container presents a concept that is different from any covered in this work. This concept is called "complete containment," which means that the reactor, turbine, generator, and all auxiliary equipment are housed within the containment structure. This, of course, has both advantages and disadvantages, as discussed in a general way in Section 7.1 and Chapters 8 and 11.

Since the BONUS vapor container (see Fig. 7.35, ref. 42) houses the entire plant, its physical size is large; and with this large free volume for expansion (1,590,000 ft$^3$), the maximum accident will not cause the internal pressure to exceed 5 psig (actually, 4.3 psig). In many other respects the BONUS container is similar to the HWCTR container described in the previous section. Both of these use the composite concrete and steel type of construction, with the steel primarily above grade and the concrete below grade. In BONUS the steel shell extends upward from the foundation mat. Below grade the shell is backed up on the outside by reinforced concrete; above grade, it consists of a short cylindrical section surmounted by the steel hemispherical dome. The foundation mat is reinforced concrete with individual pours joined as shown in Fig. 7.36 (ref. 42) using conventional water-stopping material. The junction between the concrete and steel is shown in Fig. 7.37 (ref. 42); it is below grade level and requires no special stress considerations, since it will not be subjected to thermal or other cyclic loadings.

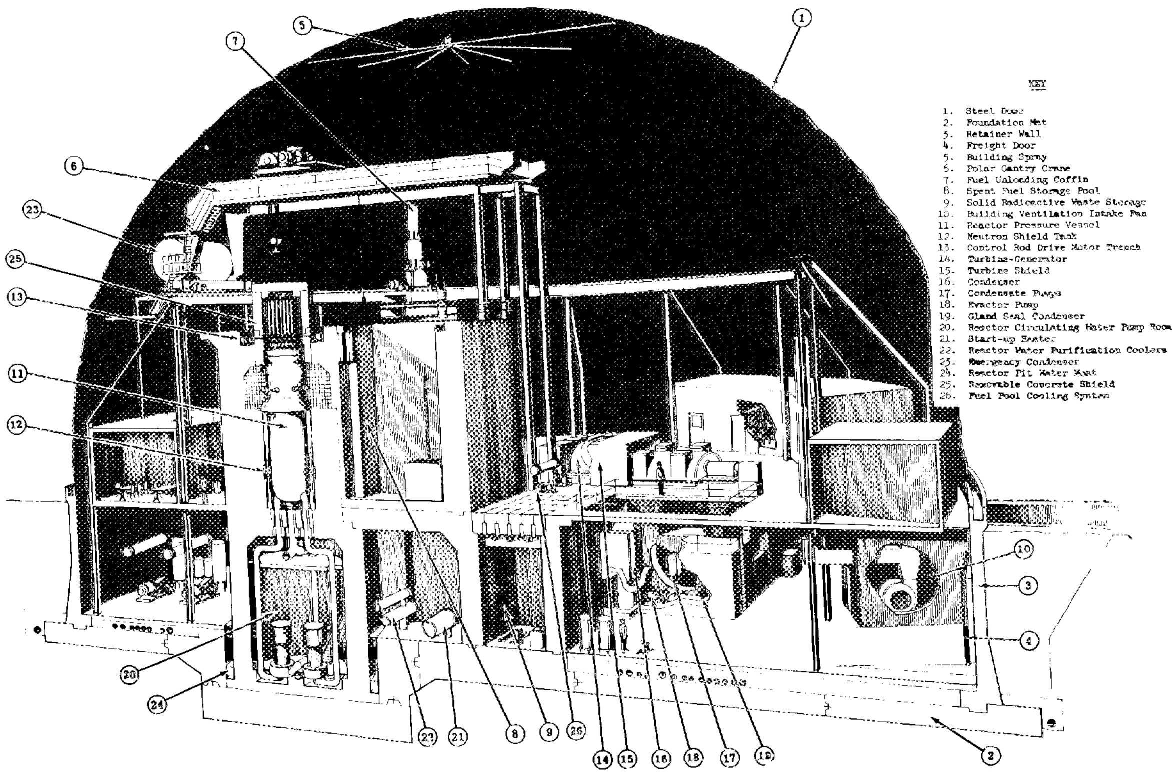



Fig. 7.35. BONUS Containment Building. (From ref. 42)

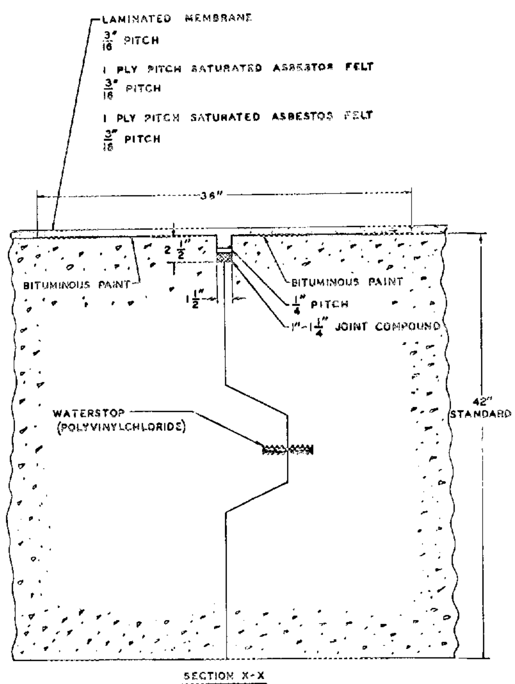
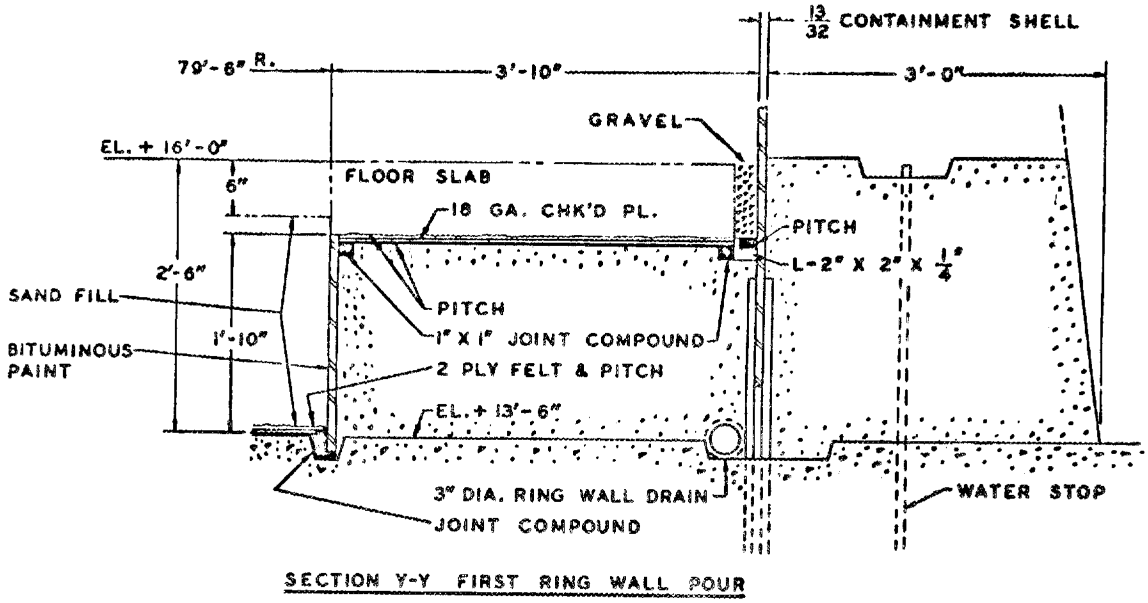



Fig. 7.36. Typical Base Slab Waterstop Installation for the BONUS Plant. (From ref. 42)



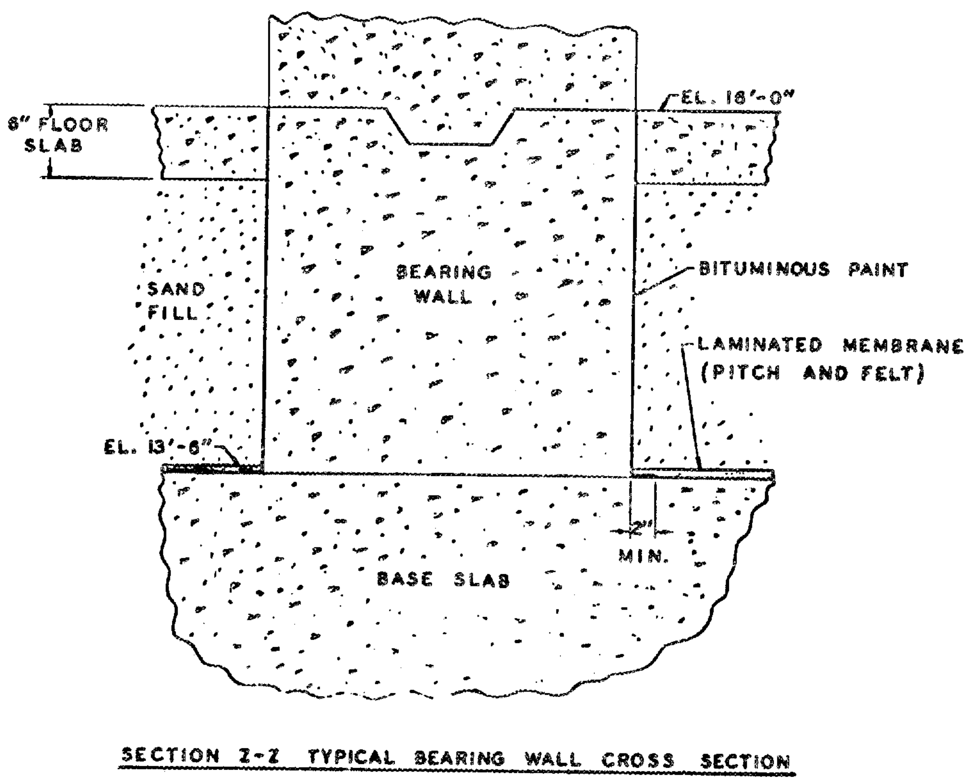




Fig. 7.37. Vapor-Barrier Membrane Details for the BONUS Plant. (From ref. 42)

In contrast to BONUS, the Piqua Nuclear Power Facility (PNPF) houses only the reactor, the fuel-handling facility, and the primary heat exchanger. The turbines and the remainder of the auxiliary and ancillary equipment are located in other buildings. Since the coolant for the PNPF is the organic material Santowax R, which has a low specific heat, the loss of primary coolant does not result in a large energy contribution to the containment. Accordingly, the internal pressure remains low under accident conditions (see Sec. 7.3.8).

Information covering name, location, thermal power, reactor type, and parties responsible for construction and operation are given in Table 7.14.

## 7.3.1 Design Parameters

Parameters such as volume, size, design pressure, and temperature and the conventional building loads are listed in Table 7.15. The BONUS Reactor is located in Puerto Rico, which has a warm climate with a high probability of occurrence of hurricanes. These two considerations dictated that provisions be made for a high wind load and a spray cooling system for the outside of the container dome to reduce the heat load to be handled by the containment air-cooling system.

The PNPF, though housed in a somewhat more conventional type of reactor container (an all-steel cylindrical shell with spherical top and ellipsoidal bottom), is designed for a pressure of only 5 psig. The interior above grade is lined with 18 in. of concrete in the cylindrical portion that decreases to 3 in. at the top of the dome. This is provided for shielding in the event of fission-product release. The reactor is below grade and is shielded by concrete in the space below the operating floor (see Fig. 7.38, ref. 43). The more conventional loads that had to be considered in the design and construction of the building are specified in the Uniform Building Code for the area.

## 7.3.2 Proof Tests

Containment buildings are subjected to the various tests described in Section 7.1 and Chapter 10. Results of these tests at BONUS and PNPF, including requirements and measured leakage rates, are listed in Table 7.16. The test procedures and methods used by these two facilities were nearly identical. The only significant difference was that the BONUS container was proof tested at −0.25 psig. Except for "Negative-Pressure Containment" (see Sec. 7.5), this is the only containment structure that is subjected to internal pressures below atmospheric. Except for this deviation the scheme used was (1) to pressurize to a low pressure (~3 psig) and look for leaks using the soap-bubble techniques, and if a leak was found, to make the necessary repairs; (2) to pressurize further to 6.25 psig (1 1/4 times the design pressure), hold for at least 1 hr, and look for stress in the vessel; and (3) to reduce the pressure to the design pressure and make a leakage-rate test for a period of three days. In both facilities the reference-vessel system (see Sec. 7.1 and Chap. 10)

Table 7.14. Low-Pressure Containment Vessels

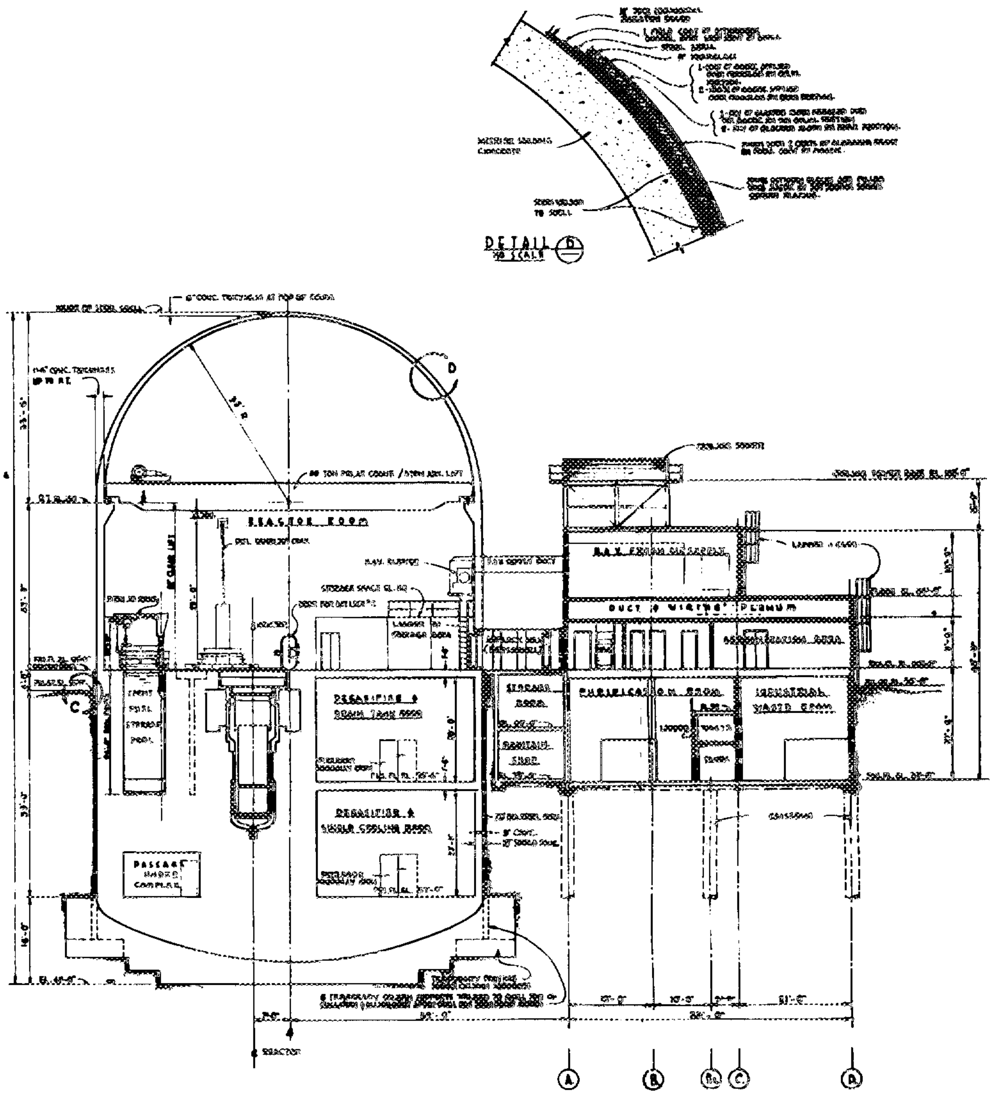
| Facility | Location | Thermal Power (Mw) | Type | Prime Contractor | Architect-Engineer | Containment Fabricator | Nuclear Equipment Supplier | Operator |
|---|---|---|---|---|---|---|---|---|
| BONUS, Boiling Water Nuclear Superheater | Punto Higuera, Puerto Rico | 50 | Boiling water with nuclear superheat | PRWRA[a] | Jackson & Moreland | Chicago Bridge and Iron | General Nuclear Engineering Corp. | PRWRA[a] |
| Piqua, Piqua Nuclear Power Facility | Piqua, Ohio | 45.5 | Organic cooled and moderated | Atomics International | Holmes & Narver | Atomics International | Atomics International | City of Piqua |

[a] Puerto Rico Water Resources Authority.

Table 7.15. Design Parameters of Low-Pressure Containment Vessels

| Facility | Shape | Free Volume ($ft^3$) | Dimensions | Design Pressure (psig) | Design Temperature (°F) | Vacuum Breakers | Wind Load | Seismic Load |
|---|---|---|---|---|---|---|---|---|
| | | $\times 10^3$ | | | | | | |
| BONUS | Vertical steel cylinder with hemispherical steel dome and flat bottom (concrete) | 1590 | 166.6 ft diam; 107.3 ft high; cylinder wall 26 ft high | 5 and −0.25 | 145 | Yes (−5 in. $H_2O$) | 150 mph | 0.2 g |
| Piqua | Vertical cylinder with spherical top and dished bottom (all steel) | 300 | 73 ft diam; 123 ft high; 65 ft above and 58 ft below grade | 5 and −0.5 | 125 | Yes (−0.5) | UBC[a] | 0.5 g vertically 0.165 g horizontally |

[a] Uniform Building Code.



SECTION A

Fig. 7.38. Piqua Nuclear Power Facility Containment Structures. (From ref. 43)

was used. Both containers met the leakage-rate specification, but at BONUS there was some difficulty with floor leaks. When the steel cylinder and dome were completed, a leakage-rate test was conducted that indicated a rate of leakage in excess of the specified value. Since this test was run before the completion of the building (i.e., the floor slab and seal had not yet been poured), it was concluded that the leakage was taking place by this path. To properly check the leakage of the steel portion of the container, the building was flooded so that the water level was above the bottom of the steel cylinder. It was feared that humidity effects because of this open water and high ambient temperature would

Table 7.16. Containment Vessel Proof Tests

| Facility | Design Leakage Requirements (%/day) | Low-Pressure Test | Strength Test Pressure (psig) | Leakage-Rate Test Pressure (psig) | Test Method | Measured Leakage Rate (%/day) |
|---|---|---|---|---|---|---|
| BONUS | 0.2 at 5 psig | Soap check | 6.25 (1 hr); 0.25 | 5 (3 days) | Reference vessel | 0.64;[a] 0.09 |
| Piqua | 0.2 per psig with a maximum of 1 at 5 psig | 3-psig soap check | 6.25 | 5 (3 days) | Reference vessel | 0.3 |

[a]The high leak rate was measured without the lower seal for the foundation mat installed; the low rate was measured with the floor flooded with water, the water to act as this seal.

be severe and difficult to correct for, but a satisfactory test was made that indicated an acceptable leakage rate.

At BONUS the integrated leakage-rate test of the entire reactor containment building was carried out at 5 psig. The frequency of integrated leakage-rate retests of the entire reactor containment building will be 1 year, 3 years, 5 years, and every 5 years thereafter. If the allowable leakage rate is exceeded during any of these tests, the testing interval sequence will revert to the beginning of the specified sequence.

### 7.3.3 Material Specifications

Material specifications and applicable codes for the containment and reactor vessels are given in Table 7.17. The code used for BONUS was API-620, rather than the ASME Code, because the ASME Code did not apply specifically to vessels with design pressures lower than 15 psig (see Chap. 2). The new Section III of the ASME Code does apply, however, to class B vessels (containment vessels) above 5-psig design pressure. With BONUS, part of the welds are hidden from visual inspection by the concrete poured around the cylindrical portion of the container. All such welds were x-rayed prior to placement of the concrete.

### 7.3.4 Penetrations

Information concerning type, size, and number of penetrations is given in Table 7.18. The means for control and closure of these penetrations in the event of an accident are described in Table 7.19.

Low-pressure building penetrations are treated in much the same manner as those of high-pressure containers. Air locks are of the double interlocking door type with gasketed closures. The large equipment doors are of the bolted and gasketed type. At the BONUS plant there are no process piping penetrations, since all primary fluids remain within the building. This leaves the ventilation closures as the only penetrations that must be depended upon to isolate the containment vessel. The PNPF is more conventional in that only the primary heat exchange system (boiler

Table 7.17. Vessel Material Specifications

| Facility | Containment Vessel Material | | | Weld Inspection | Reactor Pressure Vessel | | | |
|---|---|---|---|---|---|---|---|---|
| | Code | Specification | Thickness | | Inside Diameter (in.) | Wall Thickness (in.) | Material Specification | NDT Temperature (°F) |
| BONUS | API-620, ASA-N6 | A-201B | Side, 13/32 in.; top, 5/16 in.; concrete bottom, 3 1/2 ft | 100% x-ray[a] and API-620 | 84 | 3 1/8[b] | SA-212B | <200 at end of life |
| Piqua | | A-201B | 3/8 (max) | 100% x ray | 92 | 1 1/8–2 1/4 | SA-212B | |

[a]Welds that could not be visually observed after completion of the building were 100% x-rayed.

[b]Includes 1 1/4 in. of stainless steel cladding.

Table 7.18. Containment Structure Penetrations

| Reactor | Air Locks | Doors | Others |
|---|---|---|---|
| BONUS | Two 11-ft-diam locks with 5 × 8-ft doors | 12 × 14 ft, bolted and gasketed | 2 (5-ft) vents<br>4 (24-in.) water lines<br>1 (14-in.) water lines<br>3 (12-in.) water lines<br>2 (10-in.) vacuum relief valves<br>31 (from 1 to 10 in.) openings<br>11 electrical and control penetrations having ~115 individual leads |
| Piqua | Three locks; two with 2 1/2 × 6-ft doors, one (emergency) 2 1/2 ft ID | 16 ft ID, bolted and gasketed | 4 steam lines<br>1 relief valve<br>1 vacuum relief valve<br>4 organic coolant lines<br>2 vents<br>29 other openings<br>126 electrical penetrations |

Table 7.19. Containment Penetration Closures

| Reactor | Number and Type of Valves Per Line[a] | | | Action of Automatic Valves on Loss of Power | | Parameters Sensed to Close Automatic Valves and Number of Sensors Per Parameter | System Logic For Automatic Valves | Possible Emergency Action if Valve Does Not Close |
|---|---|---|---|---|---|---|---|---|
| | Vacuum Relief | Enclosure Ventilation | Process Lines[b] (Air, Steam, Water, etc.) | Loss of electric power | Loss of Instrument Air | | | |
| BONUS | 2 | 2 automatic (normally open) | 1 automatic | Close | Close; the instrument air is backed by a service air system; both systems must fail to cause closure | High stack activity | 2 out of 3 effluent activity monitors | Can be manually closed by activation of an electric switch in the control room or in remote operating station |
| Piqua | 1 | 2 automatic (normally open) | 1 automatic and 1 local in organic coolant lines; 1 manual and 1 local in feedwater lines; 1 automatic and 1 local in waste lines; 2 local (1 inside, 1 outside enclosure) in other lines | Close | Close | All scram parameters, 3; high stack activity, 3 | 2 out of 3; any 1 to close valves | Close locally (would take considerable time) |

[a]Automatic indicates a valve closed by instruments; manual denotes a valve operated remotely by an operator in the control room; local means a valve operated by hand at or near the valve.

[b]In this tabulation it has not been possible to take into account the pressure rating or special conditions that may apply to the system to which these lines are connected.

and superheater) is inside the containment structure. The superheated steam is taken outside the container through a high-temperature penetration.

### 7.3.5 Building Protection

Measures taken to protect the containment structures from weather and other natural hazards and from missiles generated by some credible accident to the reactor are discussed below and are listed in Table 7.20.

Table 7.20. Containment Structure Exterior and Missile Protection

| Reactor | Exterior Protection | Missile Protection |
|---|---|---|
| BONUS | Protected by paint | Turbine-generator unit is provided with a missile shield |
| Piqua | 3 in. of foam glass and resin covered by "GlasFab" and painted with aluminum | 18-in. concrete inner lining protects shell above grade level; concrete partitions, etc., below grade |

#### 7.3.5.1 External Protection

The exterior of the BONUS container is coated with commercial paint. Since the weather is not expected to be cold in this area, no thermal insulation was provided; however, the weather may be warm, so provision is made for cooling the dome by use of the external spray system. The containment shell for PNPF is fully insulated and protected, as indicated in Fig. 7.38, details C and D. These illustrations show both above- and below-grade protection.

#### 7.3.5.2 Missile Protection

The BONUS building is not protected from missiles that might be generated from the inside in the usual sense of the word. By proper placement of equipment and by concrete shields and partitions, the hazard from a missile is greatly reduced. The turbine-generator unit is provided with a missile shield. Missiles from external sources were studied for BONUS in the following manner:[42]

> "It can be proposed that a non-explosive missile pierces the containment vessel and damages the exposed equipment. In the case of the BONUS reactor there is a possibility, during a hurricane, that a missile of sufficient size and at a high velocity can penetrate the containment shell. Should this

occur there is a finite possibility that the missile may shear one of the emergency condenser steam pipe lines which are not protected.

"Operating procedures for this plant when hurricane conditions are forecast will require that the reactor be shut down, isolating valves to the emergency condenser be closed, and depressurization of the reactor be initiated by bypassing steam to the main condenser. The emergency condenser isolating valves are located below an overhanging concrete floor to protect them from missiles. Should an emergency condenser line be sheared off above these valves, no loss of steam will occur since the valves will be closed to isolate the reactor. Other critical components of the reactor plant, such as the control rod drives on the top of the reactor pressure vessel, and the turbine are provided with concrete shields which serve as radiation as well as missile shields.

"Any missile-type accident which occurs during reactor operation and which results in a core meltdown must be considered as a catastrophic-type accident such as would result during a war if an explosive missile penetrated the containment vessel. No provisions have been made for any external catastrophic accidents of this type.

"The missile accident during a hurricane has been presented since a remote possibility of this type accident occurring does exist and because it represents a type of accident wherein containment effectiveness is lost but no fission products are released."

The PNPF does not operate at high pressure (about 150 psia for the primary system), and because of this, no consideration was given to missile generation. If, however, some missile did occur, the vapor container would be protected by the various concrete structures below grade and by the 18 in. of shielding concrete above grade (see Fig. 7.38).

### 7.3.6 Moderator and Coolant

Table 7.21 gives the moderator and type and amount of coolant and its thermodynamic properties under accident conditions. The BONUS reactor is water cooled and moderated, and the PNPF uses the organic material Santowax R, which is discussed below:[42]

"The organic coolant used in the PNPF is a commercially-available organic hydrocarbon mixture consisting of the three isomers of ortho-, meta-, and para-terphenyl. The chemical formula is $C_{18}H_{14}$ and the molecular weight is 230. The ortho-terphenyl constitutes 10–15% by weight, the meta-terphenyl 55–70%, and the para-terphenyl 20–30%.

"The terphenyls are aromatic hydrocarbons, homologous to benzene, consisting of three aromatic rings linked together by means of co-valent bonds. Terphenyls are normally produced by pyrolysis of benzene at about 600°C. Terphenyl was selected

Table 7.21. Moderator and Coolant Properties at Assumed Accident Conditions

| Reactor | Reactor Type | Moderator | Coolant | Coolant Properties at Assumed Accident Conditions | | |
|---|---|---|---|---|---|---|
| | | | | Quantity (lb) | Pressure (psia) | Temperature (°F) |
| BONUS | Boiling water with nuclear superheat | $H_2O$ | $H_2O$ | | 965 | 545 |
| Piqua | Organic moderated and cooled | Santowax R | Santowax R | 76,450 | 120 | 549 |

as the moderator-coolant after an extensive testing program covering many possible hydrocarbons. The terphenyls exhibit high thermal and radiation stability, are non-corrosive, and have a low vapor pressure at reactor operating conditions.

"Under operating conditions in the PNPF, the coolant undergoes some radiation damage as it is circulated through the reactor core. This results in the formation in the coolant of various decomposition gases and higher-molecular-weight compounds. The decomposition gases, consisting primarily of hydrogen, methane, and similar materials, are removed in the coolant degasification system by a flashing and venting process. The high-molecular-weight compounds (referred to as high boilers or HB) consist primarily of long-chained polymers and are removed by vacuum distillation in the coolant purification system. The reactor will be operated with about 30% HB in the coolant."

A summary of some physical properties of Santowax R is presented in Table 7.22.[44]

## 7.3.7 Design Accident

The "maximum credible accident" is defined in Section 7.1 and is described in Sections 7.2.9 and 7.2.10 for high-pressure containment. The accident for low-pressure containment is similar. The conditions and energy releases assumed for the accident that dictate the pressure capability of the containment building are given in Table 7.23.

The BONUS accident consists of rupture of the largest coolant pipe at the bottom of the reactor, with complete discharge of water from the pressure vessel in 4 sec. The internal spray system is assumed to be inoperative. No credit is taken for heat loss to the pump room moat. The moat, as shown in Fig. 7.35, would act as a pressure-suppression system; the vapors trapped in the pump room below the reactor would bubble into the containment building through the moat. Upon doing so, steam would be condensed, and at least some of the fission products would be removed. For purposes of analysis it was assumed that the water and steam would be released to the container with no heat loss (adiabatic case) and

Table 7.22. Physical Properties of Piqua Coolant

| | |
|---|---|
| Density at 30% HB[a] content | |
| 519°F | 0.946 g/cm$^3$ |
| 575°F | 0.921 g/cm$^3$ |
| Viscosity at 30% HB | |
| 519°F | 1.03 centipoises |
| 575°F | 0.79 centipoise |
| Thermal conductivity | |
| 519°F | 0.0680 Btu/hr·ft·°F |
| 575°F | 0.0666 Btu/hr·ft·°F |
| Hydrogen density at 30% HB | |
| 519°F | $3.40 \times 10^{22}$ atoms/cm$^3$ |
| 575°F | $3.31 \times 10^{22}$ atoms/cm$^3$ |
| Melting point at 30% HB | 278°F |
| Water solubility at 510°F | |
| 120 psia | 1.4 wt % |
| 300 psia | 3.15 wt % |
| Latent heat of vaporization | |
| 519°F | 127 Btu/lb |
| 575°F | 126 Btu/lb |
| Vapor pressure at 750°F and 30% HB | 20 psia |
| Specific heat at 30% HB | |
| 519°F | 0.495 Btu/lb·°F |
| 575°F | 0.502 Btu/lb·°F |

[a]High molecular weight compounds referred to as high boilers, or HB.

with perfect mixing in the containment building. This resulted in a pressure of 4.3 psig, with an assumed building leakage rate of 0.2% per day, which persists for a period of 90 days.

The PNPF staff feel that their system, being low pressure and having a coolant with low stored energy, is not subject to the same type of accident that might befall a water-cooled reactor. The building was designed to withstand an internal pressure of 5 psig. This is the pressure (actually 4.9 psig) that might be expected if there were simultaneous failure of the steam generator and the ventilation system. This is not considered to be a credible accident. Even if this did occur, it would not in any way compromise the primary (low-pressure organic-coolant) system, so no fission products would be released to the building.

Table 7.23. Maximum Accident For Reactors With Low-Pressure Containment Vessels

| Reactor | Accident Description | Energy Sources | | Core Spray | | Building Spray Installed | Comments |
|---|---|---|---|---|---|---|---|
| | | Chemical | Stored | Installed | Used | | |
| BONUS | Total loss of primary coolant through largest credible break (16-in. line at bottom of reactor); pressure vessel empties in 4 sec | Removed by moat | 15.3 $\times 10^6$ Btu | Yes | No | Yes | Core injection system fails; the interior building spray operates 30 sec after the building becomes pressurized and removes 95% of the suspended halogens, but the pressure remains at 4.3 psig |
| Piqua | Steam boiler ruptures with ventilation system closed; this is simultaneous failure of boiler and ventilation system | | | | | | |

## 7.3.8 Fission-Product Release and Transport

The foregoing section described the event that would initiate the maximum accident and furnish the internal pressure which would cause the building contents to leak. This section deals with the remaining portion of the mca, the fission-product release and transport. Tables 7.24, 7.25, and 7.26 list the assumed releases, doses, and transport conditions following the accident.

In the safety analysis for the BONUS facility, several mechanisms are cited that would reduce the seriousness of the postulated accident. However, credit is not taken for these mitigating circumstances.

Table 7.24. Fission-Product Release in the Event of an Accident

| Reactor | Operating History | Fission-Product Release (%) | | Removal |
|---|---|---|---|---|
| | | Release to Container | Available for Release from Container | |
| BONUS | Long term at full power | 100 noble gases; 50 halogens; 1 other | 100 noble gases; 25 halogens; 0 solids | All nonvolatiles plate out in the building, half of the halogens (25% of total halogen inventory) plate out (95% of remaining halogens are washed out by building spray) |
| Piqua | Cold core (startup) | None | None | Startup accident with rapid withdrawal of control rod |
| | Full life, then shutdown for 8 hr | 100 volatiles | 100 volatiles | See discussion |

Table 7.25. Doses from Fission-Product Release from Low-Pressure Containment Structures

| Reactor | Direct Dose | Dose from Cloud | Thyroid Dose | Site Boundary | Comments |
|---|---|---|---|---|---|
| BONUS | 10 rems | | 300 rems | 0.3 mile | For direct dose, all fission products remain above the operating floor; site boundary, 0.3 mile; time for dose, 2 hr; considering no building spray, i.e., no washout of halogens |
| Piqua | <0.1 r | | | | Week at 750 ft (nearest resident) |
| | 22 rems[a] | 3.2 rems[a] | 36 rems[a] | | 1 day at 750 ft[a] |

[a]From a calculation by M. B. Biles, ref. 45.

Table 7.26. Dispersion Conditions Assumed for Accident Analysis

| Reactor | Assumed Leakage (%/day) | Height of Release (m) | Meteorological Conditions | Wind Speed (m/sec) | n | Sutton's Parameter |
|---|---|---|---|---|---|---|
| BONUS | 0.4 for entire accident (90 days) | 0 | Severe inversion | 0.87 | 0.5 | $C^2 = 0.0064$ |
| Piqua | 0.2[a] | 0 | Inversion[a] | | | |

[a]From a calculation by M. B. Biles, ref. 45.

After the BONUS primary system has ruptured, the core is assumed to melt, releasing the fission products (Table 7.24). The operation of the core injection system would have prevented this, but for purposes of this analysis this system is considered to fail. The effect of the pump room moat is likewise not considered. The following comments were taken from ref. 42:

> "In order for the steam-water mixture leaving the broken pipe to escape from the pump room, it must blow out through the flooded vent openings around the perimeter of the pump room. The pump room has been designed to withstand a momentary internal pressure of 20 psig, which would occur as a result of a large rupture in a reactor water pipe with the reactor pressurized at 950 psig. It has been calculated that, if the steam-water mixture from the reactor mixed intimately with the cold moat water that an increase of the moat water temperature from 90°F to 205°F would be sufficient to absorb all of the energy contained in the reactor water (plus heat extracted from the vessel and internals in a 30-second interval). Thus, with perfect mixing in the moat the building pressure will not rise during blowdown. Perfect mixing is not expected to occur, and therefore the vent openings will

be temporarily unflooded as the pressure in the pump room builds up and is relieved through the vent openings. After the initial pressure surge is over (approximately 4 seconds in the worst case), moat water which has been expelled will flow back into the moat. Additional water will fill the moat when the fire sprinkler heads in the reactor pump room open (80 gpm). When the building spray system is turned on, water which accumulates around the vent openings will also contribute toward filling the moat. For example, if half of the water stored in the moat is assumed to be blown out and this water is assumed to be spread evenly over a flat slab the same size as the entire basement floor, the water will flow back into the moat at 1000 gpm. With the building spray on, this rate of fill would hold constant and the moat would be resealed in about 7 minutes. With the moat vents resealed, fission products released during subsequent meltdown of the fuel will be trapped in the pump room except for minor leakages around penetrations. Calculations have shown the melting of the hottest superheater $UO_2$ fuel will not occur in less than 6 minutes, which will allow sufficient time for the moat system to essentially refill before melting can start."

The building (internal) spray is expected to operate after 30 sec and remove 75% of the halogens released to the containment building. Calculations were made, however, that do not consider the effect of this "washdown" by the spray system. These dose calculations are given in Table 7.25. The weather conditions were assumed to be those that would yield the highest doses at the boundary.

The PNPF staff considered that the mca was rapid withdrawal of the control rods and the resultant meltdown of part of the core. In this accident, 7.5% of the core fission products[44] would be released to the coolant, but this would not be a failure of the primary system. Consequently, there would be no release of fission products to the environs. This would yield a direct radiation dose rate of 0.1 mr/hr at the time of the accident to the nearest resident (750 ft).

In order to obtain a better understanding of the hazards associated with the PNPF as compared with other reactor installations, Biles,[45] Chief of Test and Power Reactor Safety Branch of the AEC, made the following comments:

"The hazards associated with a complete core meltdown have been computed, though it appears highly unlikely that such an incident would occur. Results are indicative of the maximum possible hazard. It was assumed that complete meltdown of a full life core occurs during full power operation, releasing 100% of the volatile fission products (including iodine) to the reactor building, and an internal container pressure of 1 psig exists during the entire release. 100% of the volatiles constitute 38% of the total fission product inventory. It is reasonable to assume that essentially all of the non-volatiles would be retained by the coolant, since the coolant would not be vaporized. The following tabulation represents the probable hazard at the nearest controlled area boundary (750') from direct building radiation, direct radiation from the cloud, and the total integrated thyroid dose, assuming release of the cloud to the atmosphere occurs only through normal leakage from the building. Inversion conditions were assumed.

Total Integrated Dose Rates at 750 [ft] Resulting from Core Meltdown Accident

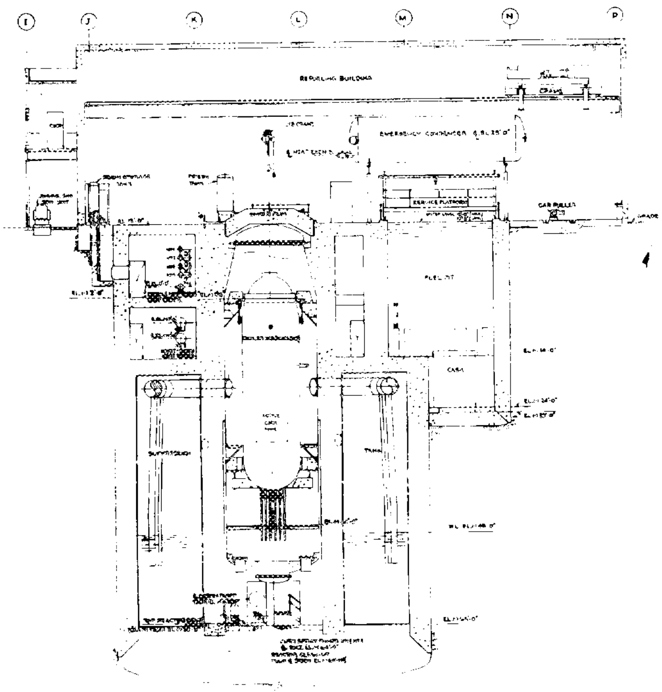
| Exposure Time | Direct Building Radiation | Direct Cloud Radiation | Thyroid Dose |
|---|---|---|---|
| 1 day | 22 rem | 3.2 rem | 36 rem |
| 1 week | 44 rem | 6.8 rem | Not applicable |

"Examination of the above results indicate[s] that this hypothetical worst possible situation probably would not result in injurious dose rates, particularly in view of the opportunity for evacuation from the nearby areas. Any credible accidents would cause substantially lower exposure possibilities."

## 7.4 PRESSURE-SUPPRESSION CONTAINMENT

In an effort to reduce the cost of containment, the concept of pressure suppression has been employed with water-cooled reactors. In principle, this technique is especially suited to water-cooled reactors, since the major portion of the energy released upon occurrence of an mca is in the form of saturated steam, which may be removed by condensation and thereby greatly reduce the final pressure to be withstood by the containing building. This scheme, shown in Fig. 7.4, uses the "dry well" and vent piping to direct the steam that is released into the water of the suppression pool, where the steam is condensed and fission products may be partially removed.

The Humboldt Bay reactor (Fig. 7.39, ref. 28) has used this method of containment, and the owner-operator, Pacific Gas and Electric Company (PG&E), has done considerable development work in the area of pressure suppression. With information from tests by PG&E, the Army built and put into operation the SM-1A reactor (Fig. 7.40, ref. 46), which uses the pressure-suppression concept. The Bodega Bay reactor (Fig. 7.41, ref. 47), which has been proposed by PG&E, will also use this method of containment. These facilities are described in Table 7.27.



SECTION E-E

Fig. 7.39. Humboldt Bay Containment Structure. (From ref. 28)

Table 7.27. Pressure-Suppression Containment Systems

| Reactor | Location | Thermal Power (Mw) | Type | Prime Contractor | Architect-Engineer | Containment Fabricator | Nuclear Equipment Supplier | Operator |
|---|---|---|---|---|---|---|---|---|
| Humboldt Bay Power Plant Unit No. 3 | Eureka, Calif. | 165 | Boiling water (natural circulation) | Bechtel | Bechtel | Suppression Chamber, Bechtel Dry Well, Chicago Bridge and Iron | General Electric Co. | Pacific Gas and Electric Co. |
| SM-1A, stationary, medium power | Fort Greely, Alaska | 20.2 | Pressurized water | | Peter Kiewet | Chicago Bridge and Iron | | U.S. Army |
| Bodega Bay Atomic Park Unit No. 1 | Bodega Bay, Calif. | 1008 | Boiling water | Pacific Gas and Electric Co. | Pacific Gas and Electric Co. | | General Electric Co. | Pacific Gas and Electric Co. |

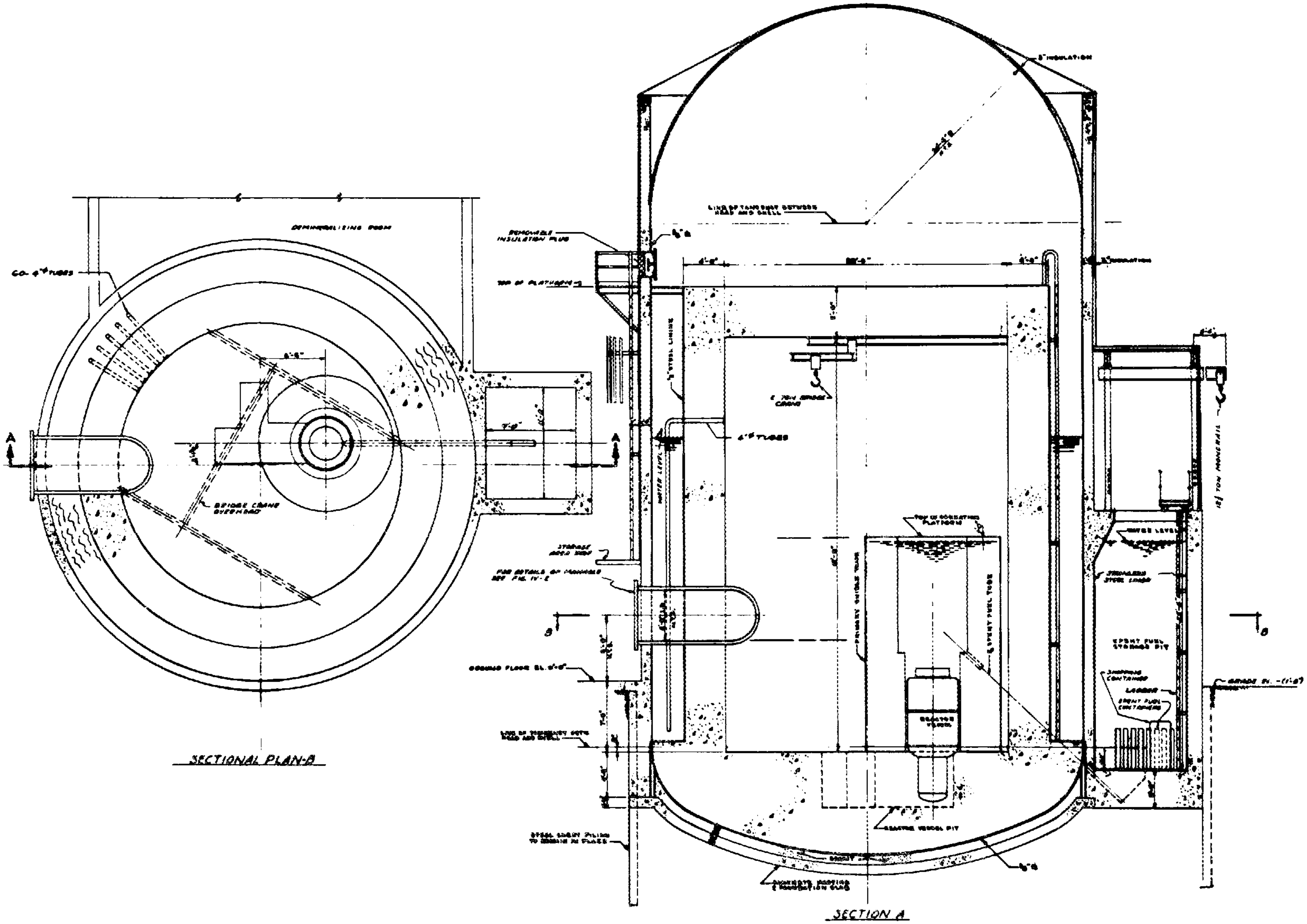



Fig. 7.40. SM-1A Container. (From ref. 46)

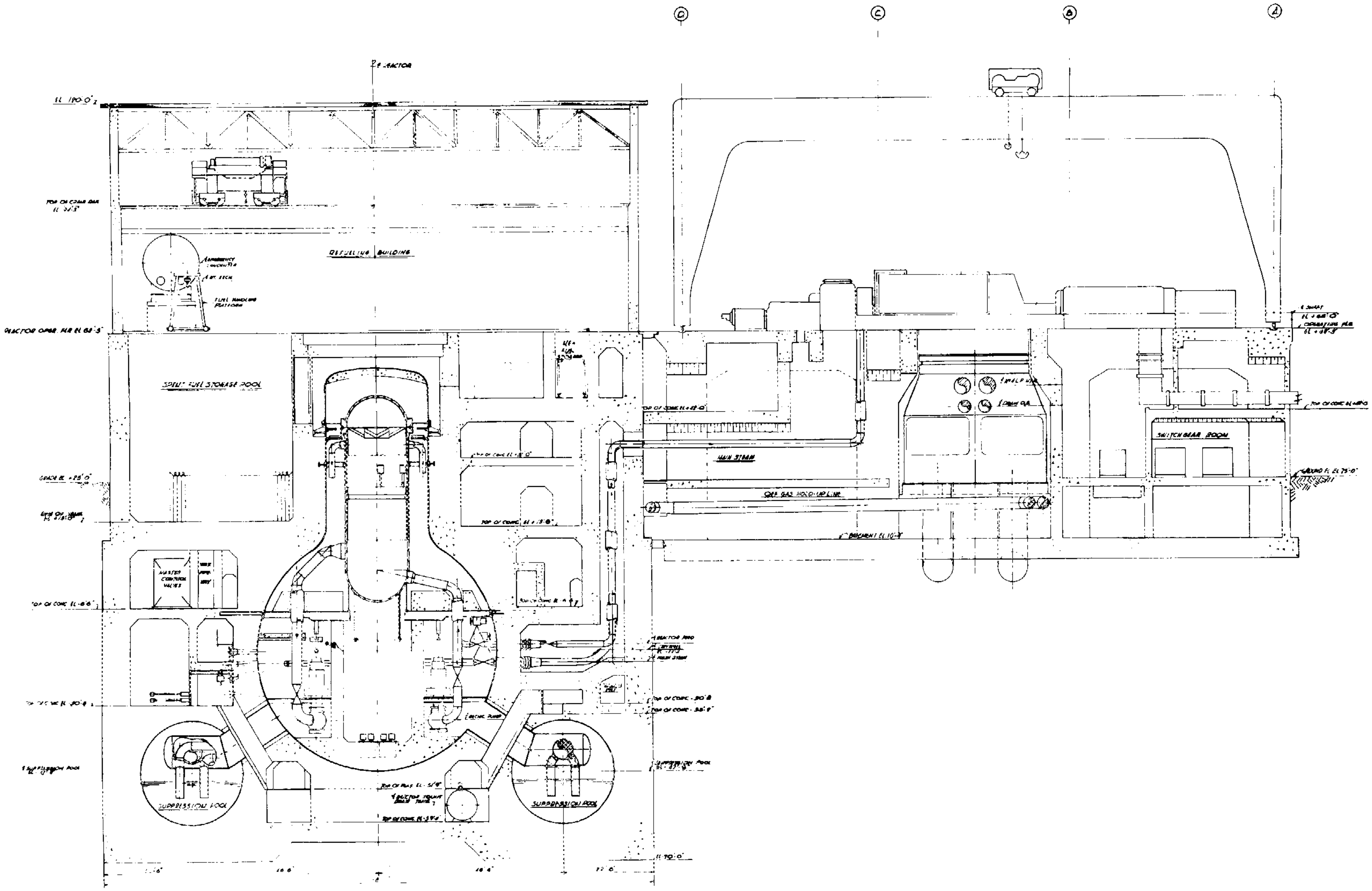



Fig. 7.41. Bodega Bay Containment Structure. (From ref. 47)

## 7.4.1 Description and Design

The system and design philosophy for pressure-suppression containment are discussed in this section. Table 7.28 lists information such as size, shape, dimensions, etc., as well as design loads from temperature changes, pressure, and conventional loadings.

One of the reasons this system may be less costly (see Chap. 11) than others for a given reactor is that the system is physically smaller. The dry well is only to be large enough to conveniently house the reactor vessel and a minimum amount of auxiliary equipment. This necessarily makes the free volume of the dry well small, and as a result the pressure peak to be withstood therein may be large. This may be seen by entries in Table 7.28 for both the Humboldt Bay unit and the SM-1A. It is thought that the "small" container with high-pressure capability may be cheaper to build and erect than the large sphere that would be necessary for the more generally accepted high-pressure containment. One serious problem with this approach is that the primary system piping penetrates the primary accident containment structure, and thus the closure of block valves in the primary coolant lines must be depended upon to effect containment for the accident. In the Humboldt Bay plant these valves and their associated piping are located in the piping tunnel leading to the turbine and both are open to atmosphere.

The suppression chamber is the second portion of this scheme. This provides a water pool into which the steam may be directed through mixing nozzles (SM-1A) or through straight pipes into a baffled suppression pool (Humboldt Bay). This container must also have high integrity against leakage, but its pressure capability is quite low (10 psig for Humboldt Bay), as may be seen in Table 7.28. The effectiveness of this concept depends on the presence of water in the suppression pool; but, even if the water level fell as much as 6 in. below the ends of the vent pipes, the condensation of steam would take place with good efficiency.[28]

In the previous discussion the dry-well vessel was referred to as the primary accident container; this implies that there is also a secondary accident container. In the case of Humboldt Bay this is true, inasmuch as the refueling building is also an independent containment scheme. This is a negative-pressure or "ORR" type of container (see Sec. 8.5) in which, under accident conditions, the building is kept at a slightly negative pressure so that air will leak into the building. Under normal operating conditions the building is unlimited in a conventional manner, and upon the occurrence of an incident the negative pressure system is put into operation. The exhausted air is then filtered, scrubbed, and discharged up a 250-ft stack.

Experiments were conducted by PG&E at their facility at Moss Landing, California,[28] on the suppression of pressure by water pools. The experimental system consisted of a full-scale segment of the Humboldt Bay system. Experiments done at this facility indicated that the dry well of the Humboldt Bay Reactor might expect a maximum pressure peak of 36 psig, instead of the design pressure of 72 psig. The dry-well vessel is conservatively designed because of the limited experimental information.

The steam temperature in the dry well of the Humboldt Bay plant may be expected to reach 575°F for a very short (30-sec) period subsequent

Table 7.28. Design Parameters for Pressure-Suppression Containment

| Reactor | Shape | Free Volume ($ft^3$) | Dimensions (ft) | Design Pressure (psig) | Temperature (°F) | Negative Pressure Protection | Wind Load | Snow Load | Seismic Load |
|---|---|---|---|---|---|---|---|---|---|
| | | $\times 10^3$ | | | | | | | |
| Humboldt Bay | Dry well: vertical steel cylinder | 12.5 | 17.5 diameter<br>67.5 high | 72 and −0.5 | 290 | Yes | (a) | (a) | 0.25 g |
| | Suppression chamber: vertical cylinderical annulus; concrete lined with steel | 33.4 (above water) | 26.5 ID<br>51.5 OD<br>49 high<br>(18 of water) | 10 | | Yes | (a) | (a) | 0.25 g |
| | Refueling building: rectangular parallelepiped; reinforced concrete | 192 | 103 long<br>43 wide<br>34 high | ±7 in. $H_2O$ | DNA | Yes | According to California Administrative Code and Uniform Building Code | | 0.25 g |
| SM-1A | Dry well: vertical cylinder; concrete lined with steel | 21.5 | 28 diameter<br>41 high | 120 | | Yes | | | 0.5 g (ver.); 0.165 g (hor.); |
| | Suppression chamber: vertical steel cylinder with spherical top and ellipsoidal bottom | 26.5 (above water) | 43 diameter<br>79 high | 15 | | Between dry well and suppression chamber | | | 0.5 g (ver.); 0.165 g (hor.) |
| Bodega Bay | Dry well: sphere surmounted by a vertical cylinder (a pear shape) of steel backed up by reinforced concrete | 115 | 60.0 diameter sphere<br>26.0 diameter for cylinder<br>100.0 overall height | 62 | 280 | | (a) | (a) | |
| | Suppression chamber<br>Steel torus | 80.2 (above water)<br>(62,400 of $H_2O$) | 93 major diameter<br>26 cross-section diameter | 35 | 150 | | (a) | (a) | |
| | Refueling building: rectangular reinforced concrete | | | | | | | | |

[a]Does not apply.

to the mca (called mcoa, maximum credible operating accident). The steel of the vessel would not show any changes in properties at this temperature, and the concrete surrounding the container is reported[48] to exhibit good structural properties up to 600°F, so no difficulties are expected as a result of the 575°F temperature transient. Other design considerations are listed in Table 7.28.

### 7.4.2 Tests

The static leak and pressure tests performed on the dry-well and suppression chambers are similar to the tests on the high- and low-pressure containers (Table 7.29). These tests have been adequately described in the appropriate sections. It is worth noting that the Humboldt Bay staff is studying the design of a system that will allow continuous monitoring for containment leakage.

Table 7.29. Containment Structure Proof Tests

| Reactor | Design Leakage Requirement (%/day) | Low-Pressure Test | Strength Test Pressure (psig) | Leakage-Rate Test Pressure (psig) | Test Method | Measured Leakage (%/day) |
|---|---|---|---|---|---|---|
| Humboldt Bay | Dry well: 0.1 at 72 psig | 5-psig soap check | 90 | 72 (24 hr) | Reference vessel | <0.1 at 72 psig |
| | Suppression chamber: 1.0 at 10 psig | 5-psig soap check | 20.3 | 10 | Reference vessel | <1.0 at 10 psig |
| | Refueling building: 100 at −1/4 in. $H_2O$ | | | At −1/4 in. $H_2O$ the system should exhaust 134 cfm to yield 100% of building volume per day | | 100 at −1/4 in. $H_2O$ |
| SM-1A | Suppression chamber: 0.016 at 5 psig | 2-psig soap check | 18.75 | 4 1/2 check of all welds and penetrations (no leak detected) | Pressurized with helium and probed with leak detector | |
| Bodega Bay | Dry well: 0.5 at 62 psig | 5-psig soap check | 71.25 | 62 | Reference vessel | |
| | Suppression chamber: 0.5 at 35 psig | 5-psig soap check | 40.25 | 35 | Reference vessel | |
| | Refueling building: 100 inleakage at −1/4 in. $H_2O$ | | | | | |

### 7.4.3 Codes

Table 7.30 lists materials of construction and the code that governs the construction, as well as information on the reactor-vessel material.

### 7.4.4 Penetrations

The type and number of penetrations are listed in Table 7.31, and the penetration closures are described in Table 7.32. With few exceptions, the penetrations of the high-integrity vapor barrier are similar to those discussed in Section 7.2.4 for high-pressure containers. The penetrations for the Humboldt Bay refueling building are included and are discussed in Section 7.5, since this portion of the Humboldt Bay containment system falls under the negative-pressure group.

One notable variation from the norm in Humboldt Bay dry-well penetrations is shown in Fig. 7.30, which is a diagram of the main steam penetration; it may be seen that the steam line itself is enclosed by a guard pipe welded to the body of the block valve. This affords protection in the event of a break in the steam line between the containment wall and the isolation valves. If this were to occur, the released steam would be directed into the dry well and subsequently to the suppression chamber. If, however, the break occurred on the turbine side of the block valves (there are two in series), these valves would be depended upon to limit the amount of steam released to the piping tunnel between the vapor container and the turbine. Both the piping tunnel and the turbine are directly open to the atmosphere.

A proposal was made to use the pressure-suppression system in conjunction with a boiling-water reactor for the City of Los Angeles reactor, but this was withdrawn when the ACRS indicated that further modifications were needed to render the plant suitable for the site. The seven suggested modifications were stated in a letter issued by the ACRS dated November 14, 1962, as follows:

1. A vapor suppression system that includes separation of primary and secondary containment,
2. A secondary containment building to withstand a pressure of 5 psig and having a leakage rate of 0.5% per day or less,
3. A method for rapid detection of fission-product release from fuel element failures,
4. A steam-line tunnel integral with the secondary containment,
5. Double isolation valves of proven type, at least one to be a turbine stop valve protected by steam strainers,
6. Holdup or detention capability for the anticipated noble gas releases to ensure that no significant environmental exposures result,
7. Turbine housing provided with controlled ventilation to filter and stack.

Table 7.30. Vessel Material Specifications

| Facility | Containment Vessel Material | | | Weld Inspection | Code Stamp | Reactor Pressure Vessel | | | |
|---|---|---|---|---|---|---|---|---|---|
| | Code | Specification | Thickness | | | Inside Diameter (in.) | Wall Thickness (in.) | Material | NDT Temperature (°F) |
| Humboldt Bay | | | | | | | | | |
| Dry well | ASME | SA-201-B, SA-300 | 5/8 in. (min) | 100% x-rayed | Yes | 120 | 4 5/16 (+1/4 SS clad) | SA-302-B | 10 in high flux region; 30 in rest |
| | ACI | Concrete | 4 to 7 ft | (a) | (a) | | | | |
| Suppression chamber | API | | 3/16 in. (liner) | 10% x-rayed | (a) | | | | |
| | ACI | Concrete | ~4 ft | (a) | (a) | | | | |
| Refueling building | ACI | Concrete | 1 ft minimum (for shielding considerations) | | | | | | |
| SM-1A | | | | | | | | | |
| Dry well | ACI-318-56 | Concrete | 3 1/2 ft (1/8 in. steel outside) | (a) | (a) | 45 1/2 | 2 5/8 (+1/4 304 SS clad) | A-212-B | 370 after 20-yr lifetime |
| Suppression chamber | API-620 | A-201-B, A-300 | 1/2 in. | 100% x-rayed | No | | | | |
| Bodega Bay | | | | | | | | | |
| Dry well | ASME Code, Sec. VIII | A-201-B, A-300 | 1 5/16 in. (max) | 100% x-rayed | Yes | 181 | ~6 (+1/4 304 SS clad) | SA-302-B | |
| Suppression chamber | ASME Code, Sec. VIII | A-201-B, A-300 | 1 in. (max) | 100% x-rayed | Yes | | | | |
| Refueling building | (b) | (b) | (b) | (a) | (a) | | | | |

[a] Does not apply

[b] Data not available

Table 7.31. Containment Structure Penetrations

| Facility | Air Locks | Access Openings | Others |
|---|---|---|---|
| Humboldt Bay Dry well | | Two flanged openings: 14 ft at top, 6 ft at bottom | 6 (40 in.) dry-well vents[a]<br>1 (24 in.) steam line<br>4 (24 in.) hydraulic lines<br>1 (20 in.) feedwater line<br>1 (20 in.) vacuum relief valve<br>2 (18 in.) shutdown cooling lines<br>49 (1 to 16 in.) others<br>5 electrical lines |
| Suppression chamber | | Two 30-in. manholes | 2 (20 in.) vacuum relief valves<br>4 (10 in.) discharge lines<br>12 (1 to 6 in.) others |
| Refueling building | Two with 2-ft × 1 1/2-in. × 5 1/2-ft marine-type doors | Railway door | Many |
| SM-1A Dry well | From outside (see Fig. 7.40) to dry well through the space between dams, to be water-filled when operating | | Various pipes and conduits |
| Suppression chamber | | 30-in. manhole | |

[a]Part of containment system.

## 7.4.5 Containment Protection

The protection measures provided for these containers are listed in Table 7.33. The Humboldt Bay primary container is housed in a concrete building and needs no further external protection. The dry-well vessel is designed to handle the missile and jet forces to which it might be subjected.

The SM-1A steel container is protected in the cylindrical portion by an external concrete wall (Fig. 7.40) and on the top by insulation and conventional roofing materials. The SM-1A is protected from missiles by the dry-well container walls, which are of reinforced concrete at least 4 ft thick. It is thought that no conceivable missile could damage this barrier to the extent of voiding the container.

The Bodega Bay dry-well vessel is backed up by reinforced concrete that has adequate strength to withstand any jet or shock loadings that might accompany the mca.

## 7.4.6 Coolant Properties

Table 7.34 lists the type of reactor and amount of coolant and the thermodynamic conditions of the coolant under normal operating conditions. Coolant and moderator for the SM-1A and Humboldt Bay are the same material, water.

Table 7.32. Containment Penetration Closures

| Reactor | Number and Type of Valves Per Line[a] | | | Action of Automatic Valves on Loss of Power | | Parameters Sensed to Close Automatic Valves And Number of Sensors Per Parameter | System Logic For Automatic Valves | Possible Emergency Action If Valve Does Not Close |
|---|---|---|---|---|---|---|---|---|
| | Relief Vacuum | Enclosure Ventilation | Process Lines[b] (Air, Steam, Water, etc.) | Loss of Electric Power | Loss of Instrument Air | | | |
| Humboldt Bay Pressure suppression | 1 | 2 (normally closed) | 2 valves at primary containment (both motor operated if outlet lines; 1 motor operated and 1 check if inlet line) | If loss is greater than 3 sec duration, valves close | Close | Reactor low water level, 4; dry-well high pressure, 4; main steam line break, 4; auxiliary power low voltage, 1; high pressure in the refueling building, 2; high building activity, 2 | 2 out of 4 to close all valves | Close by remote manual operation |
| Refueling building | 0 | 2 (normally open) automatic closure | (c) | Close | (c) | Reactor low water level, 4; dry-well high pressure, 4; main steam line break, 4; auxiliary power low voltage, 1; high pressure in the refueling building, 2; high building activity, 2 | 1 out of 2 | Close by remote manual operation |
| SM-1A[d] | | | | | | | | |
| Bodega Bay[d] | | | | | | | | |

[a]Automatic indicates a valve closed by instruments. Manual denotes a valve operated remotely by an operator in the control room.

[b]In this tabulation it has not been possible to take the pressure rating or special conditions that may apply to which these lines are connected.

[c]Does not apply.

[d]Data not available.

Table 7.33. Containment Structure Exterior and Missile Protection

| | Exterior Protection | Missile Protection |
|---|---|---|
| Humboldt Bay | | |
| Dry well | (a) | No credible missile will penetrate the dry-well liner; piping runs are placed so that they cannot become missiles dangerous to the liner plate |
| Suppression chamber | (a) | (a) |
| Refueling building | Concrete building | (a) |
| SM-1A | | |
| Dry well | | Missile protection is provided by at least 4 ft of reinforced concrete; access has a 2 1/2-in. steel door |
| Suppression chamber | Steel building is enclosed in concrete up to the bend line; from there it is covered by insulation and paint | (a) |

[a]Does not apply.

Table 7.34. Moderator and Coolant Properties at Assumed Accident Conditions

| Reactor | Type of Reactor | Moderator | Coolant | Coolant Properties at Assumed Accident Conditions | | |
|---|---|---|---|---|---|---|
| | | | | Quantity | Temperature (°F) | Pressure (psia) |
| Humboldt Bay | BWR | $H_2O$ | $H_2O$ | 83,000 | 575 | 1265 |
| SM-1A | PWR | $H_2O$ | $H_2O$[a] | 7,870 | 443 | 1215 |
| Bodega Bay | BWR | $H_2O$ | $H_2O$ | 294,000 | 575 | 1265 |

[a]Primary and secondary.

### 7.4.7 Accidents

The accidents that determine the design pressure and the temperature conditions within the containment structures are described in Table 7.35.

For the Humboldt Bay plant, a rupture of the main steam line and discharge of all reactor water through this line are postulated, and it is assumed that the reactor will have been operating at 230 Mw(t) and 1250 psig prior to the accident. The core spray system would preclude

Table 7.35. Design Accident

| Reactor | Description of Accident | Energy Release (Btu) | | | Core Spray | Comment |
|---|---|---|---|---|---|---|
| | | Nuclear | Chemical | Stored | | |
| | | | | $\times 10^6$ | | |
| Humboldt Bay | Loss of primary coolant through the largest credible rupture within the dry well (12 in. Sched.-80 pipe, single-ended system) | 0 | 0 | 45 | Yes | Core spray activates at a primary system pressure of 150 psig; core spray is partially ineffective and allows 50% of the core to melt |
| SM-1A | Same as SM-1 [two cases analyzed: case 1 results in the highest pressure in the dry well; case 2 results in highest pressure in suppression chamber (building); the worst of both accidents was taken as the design requirement] | 0 | 0 | 4.9 | No | |
| Bodega Bay | Loss of coolant from an instantaneous rupture of any pipe connected to the reactor vessel (area of break, 6.4 ft$^2$; loss of coolant in 8 sec; double-ended break) | 0 | 0 | 150 | Yes | Core spray activates at a primary system pressure of 150 psig; core spray is partially ineffective and allows 50% of the core to melt |

fuel melting, but for the sake of analysis this system was assumed to fail. (The fuel-handling accident was also analyzed for Humboldt Bay, see Sec. 7.5).

The reactor system for the SM-1A is similar to that for the SM-1 (see Sec. 7.2), the major difference in the two installations being the containment scheme. The sequence of events leading to the mca is the same as for the SM-1, as shown graphically in Fig. 7.42 (ref. 46). Since there are two parts (the dry well and suppression chamber) to the SM-1A containment scheme, both were analyzed for the highest possible pressure. A postulated accident identical to that for the SM-1 (i.e., the secondary relief valve fails to open and allows the system pressure to increase to 1500 psig) was found to result in the highest pressure in the dry well. In a second case, it was assumed that the secondary relief valve opened as it should and allowed the pressure in the outer container, the suppression chamber, to reach its maximum value. The two parts of the containment system were designed to contain the highest pressure from either maximum accident.

### 7.4.8 Activity Release

Data pertinent to the release of radioactivity after the occurrence of the maximum accident are given in Tables 7.36 through 7.40. For Humboldt Bay, two accidents were analyzed: the mcca and the fuel-handling accident. Since the fuel-handling accident would take place in the

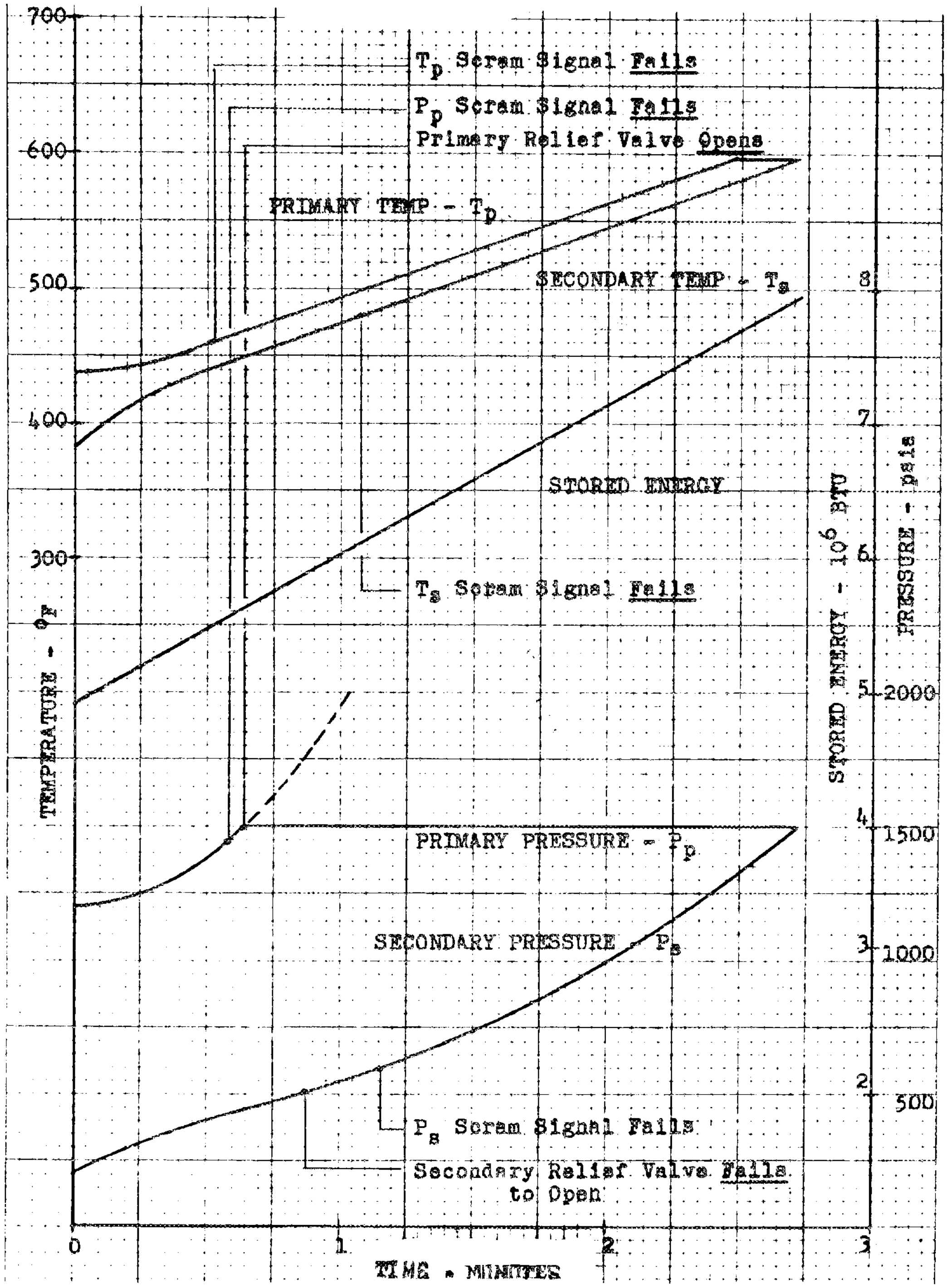



Fig. 7.42. Sequence of Events Leading to mca for the SM-1A. (From ref. 46)

Table 7.36. Fission-Product Release in the Event of an Accident

| Reactor | Operating History | Fission-Product Release to Container (%) | | Fission-Product Release from Container | Comment |
|---|---|---|---|---|---|
| Humboldt Bay | Long-term operation at 230 Mw(t) | 100 noble gases<br>25 halogens<br>15 volatile solids<br>0.3 other | To dry well | | |
| | | 100 noble gases<br>12 1/2 halogens<br>7 1/2 volatile solids<br>0.15 other | To suppression chamber pool | | |
| | | 100 noble gases<br>0.01 halogens<br>0.01 other | To suppression chamber vapor space | See Table 7.37 | These figures are for the maximum credible operating accident (mcoa), which would result in very small doses, so the fuel handling accident is also analyzed (see Tables 7.49 and 7.51) |
| SM-1A | 1.5 yr at 20 Mw(t) | 1% of all fission products | | $1.2 \times 10^{-6}$%/hr | |
| Bodega Bay | Long-term operation at 1008 Mw(t) | | | | See discussion and Fig. 7.43; see also Tables 7.49 and 7.51 for fuel-handling accident |

Table 7.37. Humboldt Bay Release Rates from Stack Following Maximum Credible Operating Accident

| Time After Event | Release Rates (μc/sec) | | | |
|---|---|---|---|---|
| | Noble Gases | Halogens | Volatile Solids | Other Solids |
| 1 hr | 7,900 | 15 | 5 | 1 |
| 3 hr | 60,000 | 60 | 21 | 4 |
| 10 hr | 240,000 | 160 | 50 | 11 |
| 1 day | 380,000 | 230 | 90 | 18 |
| 3 days | 340,000 | 170 | 80 | 19 |
| 10 days | 1,400 | 1 | 1 | 0.1 |

refueling building, where the air is controlled by negative pressure, this falls in the category of a negative-pressure system (see Sec. 7.5).

The mcoa of the Humboldt Bay reactor is based on release of fission products to the dry well, as indicated in Table 7.36. Then, in the dry well, half of the halogens and solids are assumed to plate out. When the dry-well contents are bubbled through the suppression pool, 99.99% of the solids and halogens are scrubbed out and retained. No credit is taken for the removal of the noble gases.[28] It is clearly seen here that the

Table 7.38. Doses from Fission-Product Release from Containment Structure

| | Direct Dose | Whole Body | Thyroid | Bone | Comments |
|---|---|---|---|---|---|
| Humboldt Bay | Table 7.39 | Mcoa<br>Unstable condition,[a] 0.5 millirem/hr at 0.6 mile<br>Stable condition,[a] 5 millirems/hr at 5 miles (elevated ground) | | | The mcoa assumes that most of the iodine and solids were scrubbed out by the refueling-building cleanup equipment; also see Tables 7.49 and 7.51 for fuel-handling accident |
| SM-1A | 220 mr, 8 hr at 40 m | 0.043 rem in 1 day | 0.086 rem in 1 day | 0.08 rem in 1 day | |
| Bodega Bay | | Unstable condition,[a] 0.024 rem | | 0.019 rem | Integrated dose at 0.6 mile (point of maximum dose) |
| | | Stable condition,[a] 0.11 rem | | 0.09 rem | Integrated dose at 3.0 miles (point of maximum dose) |

[a]See Table 7.40 for wind speeds.

Table 7.39. Direct Radiation from the Humboldt Bay Building Following the Maximum Credible Operating Accident

| Time After Event | Curies in Building | Dose Rate at 1/4 Mile (mrem/hr) |
|---|---|---|
| 1 hr | 720 | 0.02 |
| 3 hr | 5,400 | 0.1 |
| 10 hr | 21,000 | 0.5 |
| 1 day | 33,000 | 0.8 |
| 3 days | 30,000 | 0.7 |
| 10 days | 120 | 0.003 |

Table 7.40. Dispersion Conditions Assumed for Accident Analyses

| Reactor | Assumed Leakage | Height of Release (ft) | Wind Speed (mph) | Weather Condition | Dispersion Parameters | | |
|---|---|---|---|---|---|---|---|
| Humboldt Bay | Table 7.37 | 250 | 4<br>3 | Unstable<br>Stable | | | |
| SM-1A | 0.1%/day | 0 | 6.5 | Inversion | $n = 0.55$ | $C_z = 0.05$ | |
| Bodega Bay | Fig. 8.48 | 300<br>200 | 10<br>5 | Unstable<br>Stable | $n = 0.22$<br>$n = 0.3$ | $C_y = 0.6$<br>$C_y = 0.21$ | $C_z = 0.2$<br>$C_z = 0.09$ |

major contribution to the doses from this accident would be the noble gases (see Table 7.37). Table 7.38 indicates the doses that might result from the mcoa.

For the SM-1A the postulated catastrophe forfeits the effect of the entire containment scheme. The core melts and allows 10% of the fission-product inventory to escape from the building to the environs over a 2-hr period. Even when considering this severe accident, the arbitrary limits (i.e., 25 rems direct radiation and 300 rems inhalation) are not exceeded outside the limiting distances shown in Table 7.38.

Table 7.40 indicates leakage, height of release, and weather conditions influencing the downwind doses calculated. The Humboldt Bay leakage is not quoted in the usual manner. Instead of a percent per day specification, the discharge to the atmosphere is given as indicated by Table 7.37. This discharge is made through the filter-scrubber unit and up the 250-ft stack.

For Bodega Bay, it was assumed that the fission products would be released in the quantities shown in Fig. 7.43 (ref. 47). Credit was taken for the efficiency of the ventilation cleanup equipment, which would be brought into operation in the event of a major accident. The ventilation cleanup equipment was specified to retain not less than 95% of the halogen

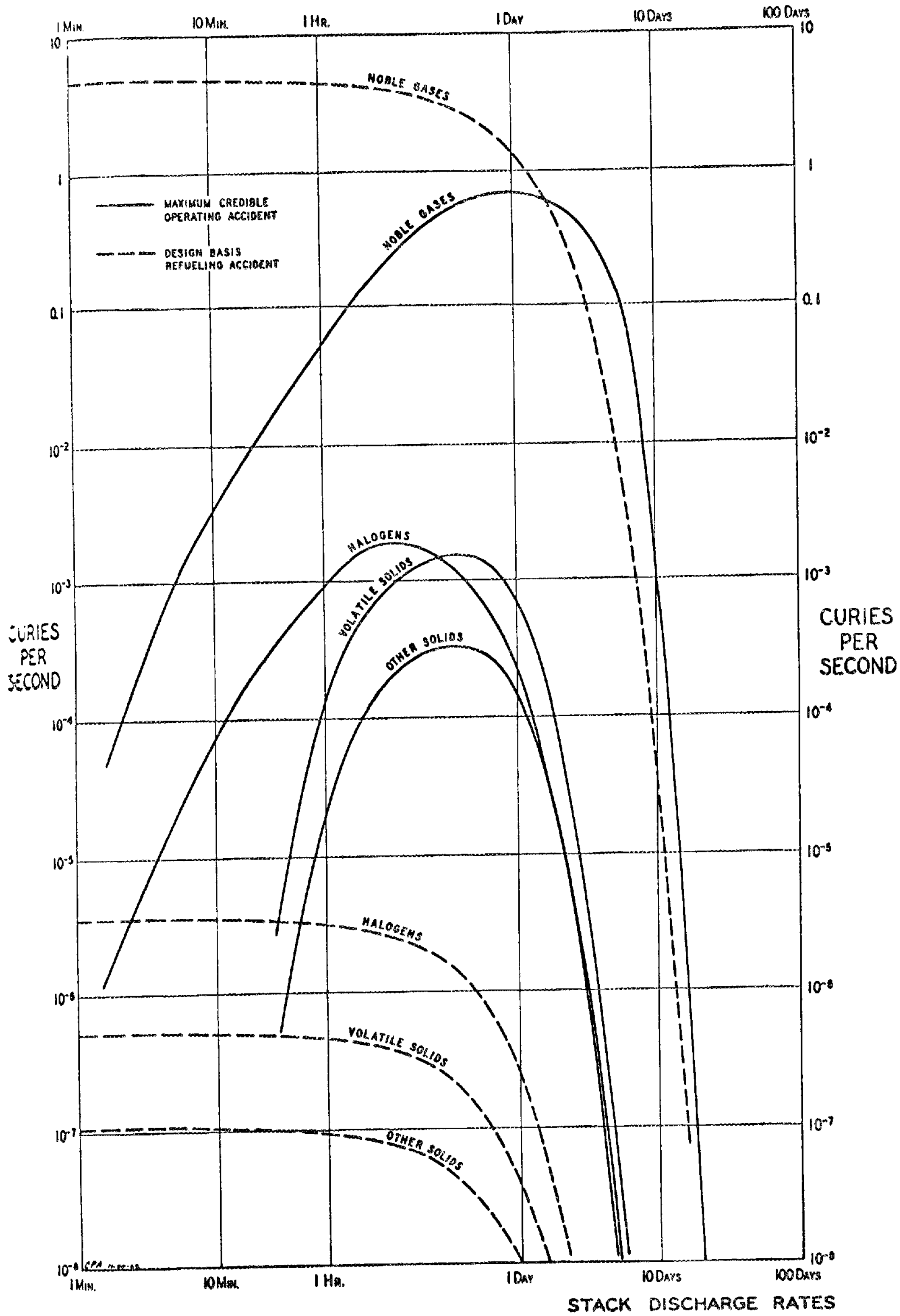



Fig. 7.43. Stack Discharge Rates of Bodega Bay Plant as a Function of Time After the Accident. (From ref. 47)

and solid fission products. Figure 7.43 shows the calculated rates of release of fission products from the stack as a function of time after the accident.

## 7.5 PRESSURE-VENTING CONTAINMENT

The system described in this section is more aptly described as a confinement system (see also Sect. 1.4.3). In this system the building atmosphere, which may contain fission products in case of an incident, is filtered and released at a controlled rate. The gases and fine particulates that do not fall out or plate out in the building are exhausted through cleanup equipment, and the effluent is finally discharged to the environs, generally from a stack.

### 7.5.1 Building Concept

In systems previously discussed, the most important single consideration was leaktightness of the containment shell, whether it be of concrete or steel. In this system, however, proper functioning depends upon the building leaking air inward in a controlled fashion under the influence of a negative building pressure, which is produced by a system of blowers. Since the building is required to leak, more conventional construction materials and methods may be utilized (see Table 7.41).

The buildings that house the four examples, i.e., the Hallam refueling building (Fig. 7.44, ref. 49), the Oak Ridge Research Reactor (Fig. 7.45, ref. 50), and the Humboldt Bay (Fig. 7.46, ref. 28) and Bodega Bay refueling buildings, are rectangular in shape and are subject to the local building codes for their construction criteria (see Table 7.42). The main departure from the conventional is the requirement with regard to pressure. Since the buildings are expected to leak in the inward direction only, they must be held at some specific negative pressure. This requirement specifies two related conditions: (1) the building must be sufficiently leaktight to produce the desired vacuum with the blower equipment that is installed, and (2) it must have strength to resist collapsing because of external pressures when the specifications of in-leakage and negative pressure are met.

### 7.5.2 Pressure and Leakage Requirements

The requirements of the building with regard to leakage are listed in Table 7.43. It can be seen by comparing the leakages in this table that there is a wide range of leaktightness requirements. The Humboldt Bay refueling building is designed for the least absolute leakage. The requirement for the ORR is at the other end of the scale. The philosophy concerning the ORR, and it is generally true for any system of this type,

Table 7.41. Pressure-Venting Containment Buildings

| Facility | Name | Location | Thermal Power (Mw) | Type | Prime Contractor | Architect-Engineer | Containment Construction | Nuclear Equipment Supplier | Operator |
|---|---|---|---|---|---|---|---|---|---|
| Hallam refueling building | Hallam Nuclear Power Facility | Hallam, Neb. | 256 | Sodium cooled | AEC | Bechtel | Peter Kiewit | AI | Consumers Public Power District |
| ORR | Oak Ridge Research Reactor | Oak Ridge, Tenn. | 30 | Pool | | | | | ORNL |
| Humboldt Bay refueling building | See Table 7.27 | | | | | | | | |
| Bodega Bay refueling building | See Table 7.27 | | | | | | | | |

Table 7.42. Design Parameters of Pressure-Venting Containment Systems

| Facility | Shape | Free Volume ($ft^3$) | Dimensions (ft) | Design Pressure (in. $H_2O$) | Loads |
|---|---|---|---|---|---|
| | | $\times 10^3$ | | | |
| Hallam refueling building | Rectangular, mill-type building | 1420 | 278 long<br>80 wide<br>75 high | −1/4 | Constructed according to local codes with regards to winds, earthquakes, etc. |
| ORR | Rectangular, mill-type building | 800 | | | Constructed according to local codes with regards to winds, earthquakes, etc. |
| Humboldt Bay refueling building | Rectangular, concrete | 192 | 103 long<br>43 wide<br>34 high | $\pm 7$ | Constructed according to local codes with regard to winds, earthquakes, etc. |
| Bodega Bay refueling building | Rectangular, concrete | | | | |

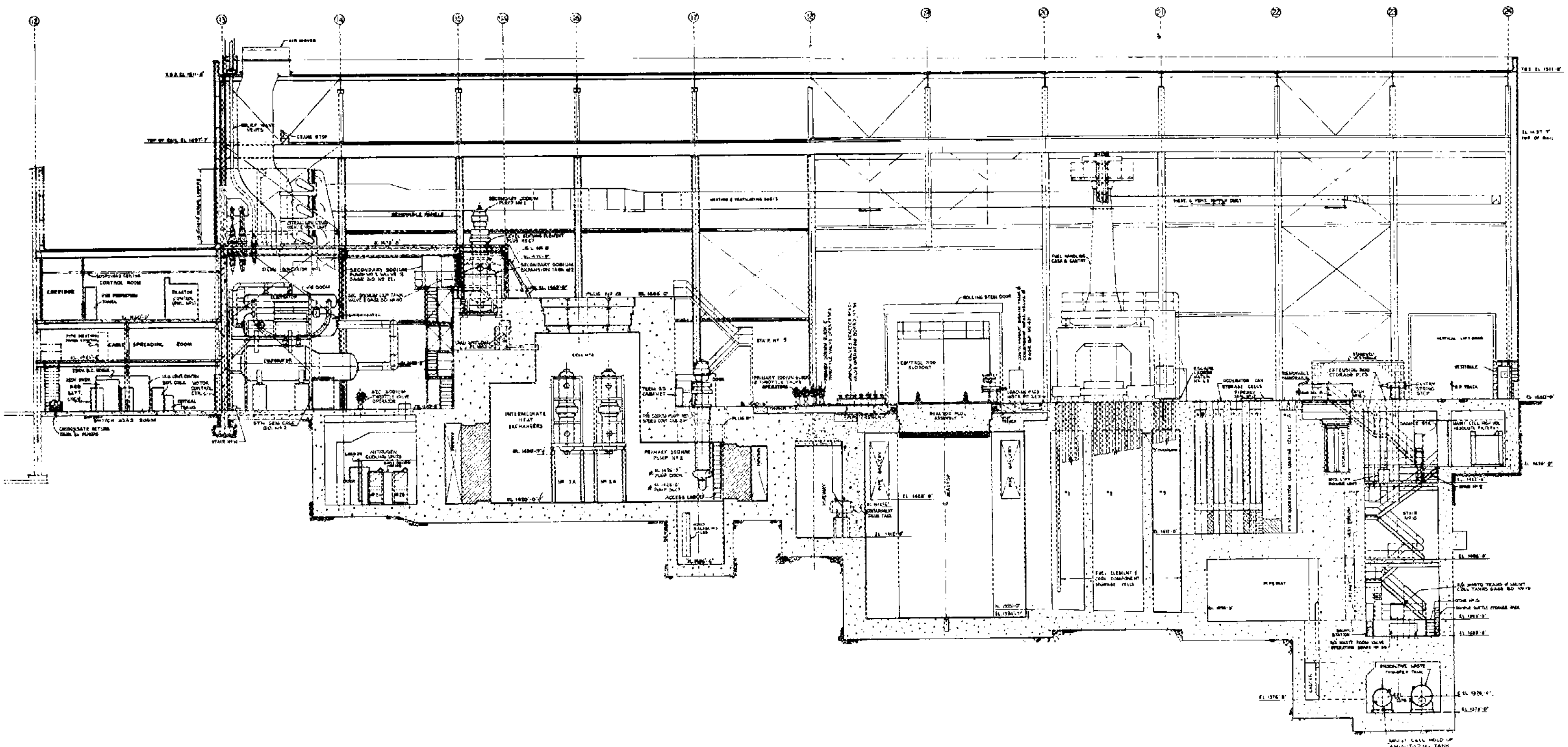


Fig. 7.44. Hallam Reactor Building. (From ref. 49)

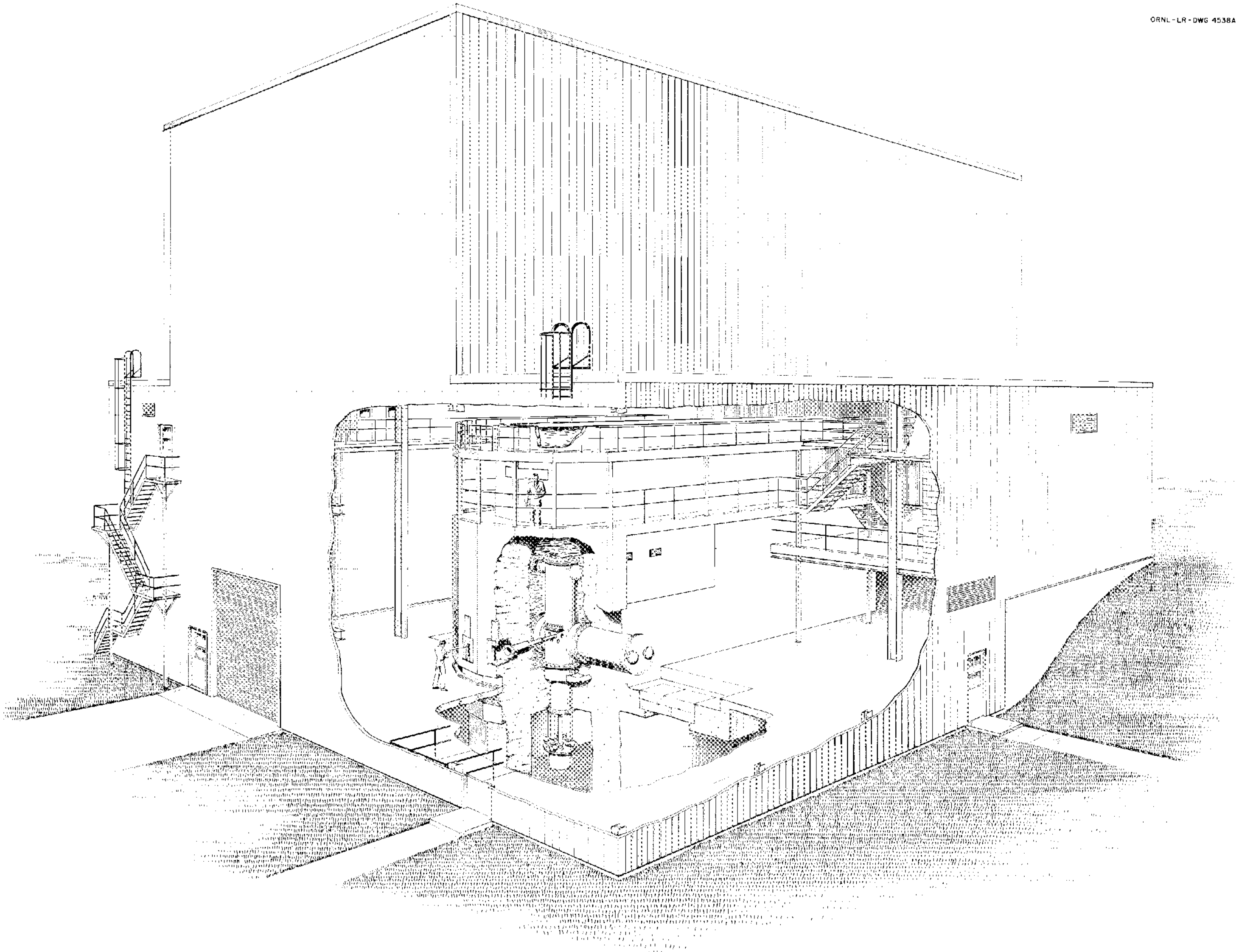



Fig. 7.45. The Oak Ridge Research Reactor Building. (From ref. 50)

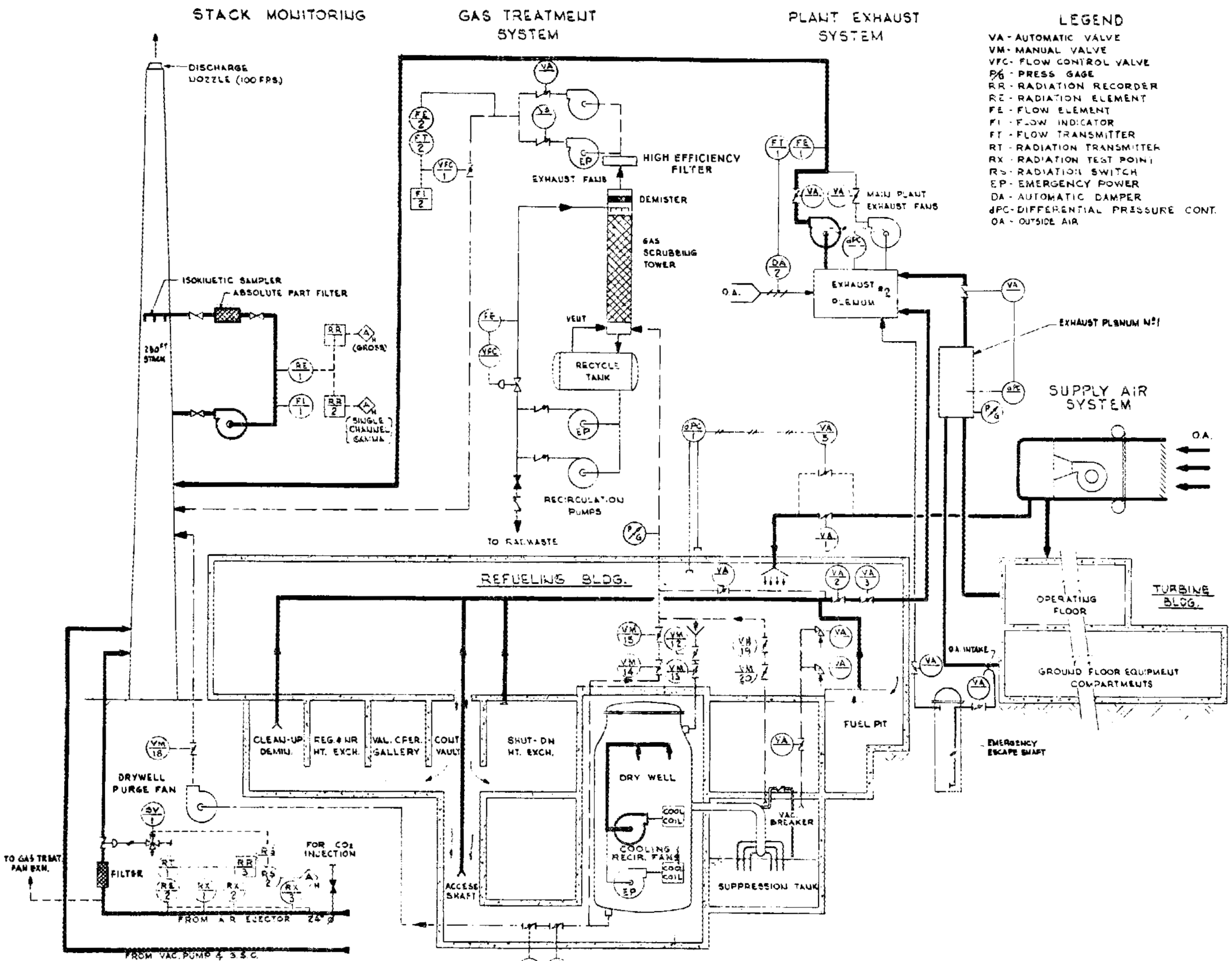



Fig. 7.46. Humboldt Bay Refueling Building Ventilation Flowsheet. (From ref. 28)

Table 7.43. Containment System Proof Tests

| Facility | Inleakage Requirement | Tests |
|---|---|---|
| Hallam refueling building | 3120 cfm at $-1/8$ in. $H_2O$ | A means of measuring leakage is provided; all building air supplies are shut off; the internal pressure is maintained at $-1/8$ in. $H_2O$; the discharge airflow is measured with a flowmeter |
| ORR | 6000 cfm at $-0.3$ in. $H_2O$ | When all openings are closed, the available building draft should result in a vacuum of 0.3 in. $H_2O$ |
| Humboldt Bay refueling building | 100%/day (134 cfm) at $-1/4$ in. $H_2O$ | The leakage is measured with an air-flow meter and a vacuum gage |
| Bodega Bay refueling building | 100%/day at $-1/4$ in. $H_2O$ | The leakage is measured with an air-flow meter and a vacuum gage |

is that[50]

> "Complete closure of all the building openings is desirable but not absolutely necessary. In fact it is absolutely necessary for sufficient air to leak into the building to replace that removed by the blowers. One of the advantages of controlled containment is that it can be made to work even though there is a hole in the building. There is qualitative evidence to indicate that containment is achieved even with one of the large truck doors open."

The qualitative evidence above refers to an experiment which was performed at the ORR site. With the blowers removing 6000 cfm of air from the building, the truck doors were opened, and at a distance of about 50 ft, a smoke generator was ignited. The smoke moved unquestionably and rapidly toward the open door. The experiment was carried out under ideal conditions (i.e., no wind), but it served to illustrate the effectiveness of this concept to maintain air movement into the building.[51]

Experiments are planned for the ORR to demonstrate the effectiveness of the containment system under a variety of weather conditions. Small ports, of the order of 1 in. in diameter, are to be installed at several locations in the building walls. These ports are to be equipped with anemometers so that the air velocity through the port may be measured at any time.

### 7.5.3 Building Construction

The ORR and Hallam buildings are similarly constructed (Table 7.44). A steel building frame is erected, and building panels are attached to the frames. The ORR building is of conventional aluminum sides and roof panels, and particular care was given to the sealing of joints. The Hallam building uses steel panels with all joints sealed, and over this there is another layer of steel panels that are thermally insulated. It has not been difficult to maintain the desired vacuum in these buildings with the specified movement of air.

Table 7.44. Containment Material Specifications

| Facility | Code | Type of Construction | Reactor Vessel: Inside Diameter (in.) | Reactor Vessel: Wall Thickness | Reactor Vessel: Metal |
|---|---|---|---|---|---|
| Hallam refueling building | UBC | Steel panels over steel framing, insulated steel panels outside | 226 | | A-204 |
| ORR | UBC | Aluminum siding panels over steel frame | | | |
| Humboldt Bay refueling building | UBC | Concrete building; walls, 1 ft thick | | See Table 7.30 | |
| Bodega Bay refueling building | UBC | | | See Table 7.30 | |

The Humboldt Bay refueling building is concrete with walls of 1 ft in minimum thickness. The building has been in use for more than a year, and no difficulty has been encountered in maintaining the desired tightness.

### 7.5.4 Penetrations

The pressure-venting systems are completely different from the containment schemes discussed in preceding sections, and the penetrations are likewise different. In particular, the requirements are less stringent (see Table 7.45) when the pressure in the containment volume is subatmospheric. Equipment and personnel doors offer potential "gross" leakage paths and are provided with closure surfaces that will seal securely. Doors are provided with closing devices, either of the conventional type,

Table 7.45. Containment Building Penetrations

| Facility | Air Locks | Doors | Penetrations |
| --- | --- | --- | --- |
| Hallam refueling building | For personnel | Railway door | Many penetrations; electrical, process, and steam are sealed by conventional methods |
| ORR | | 12 personnel doors | 12 exhaust openings from 1 to 45 $ft^2$ |
| | | 2 truck doors (12 × 16 ft) | |
| Humboldt Bay refueling building | For personnel | Railway door | Electrical and process |
| Bodega Bay refueling building[a] | | | |

[a]Data not available.

such as used on the ORR personnel doors, or motor-driven closers, such as are used with the ORR truck doors. The Hallam building has air locks for personnel entry, and the large railway door is not to be opened when the reactor is in operation. The Humboldt Bay refueling building also uses air locks with marine-type doors for personnel entry.

### 7.5.5 Building Protection

The credibility of damage by a missile from some external source is considered to be nil. Protection of the buildings from any internal missiles is provided by appropriate missile shields or by judicious placement of equipment (Table 7.46).

These are conventional buildings and rely on conventional means for corrosion protection. The ORR building is of aluminum and needs no further

Table 7.46. Containment Building External and Missile Protection

| Facility | Exterior Protection | Missile Protection |
| --- | --- | --- |
| Hallam refueling building | Paint | It is felt that no missile originating from any credible accident can violate the containment |
| ORR | Paint | It is felt that no missile originating from any credible accident can violate the containment |
| Humboldt Bay refueling building | None needed, concrete | It is felt that no missile originating from any credible accident can violate the containment |
| Bodega Bay refueling building | None needed, concrete | It is felt that no missile originating from any credible accident can violate the containment |

protection from weathering. The Hallam building is steel and is painted. The Humboldt Bay refueling building is of concrete and needs no further protection.

### 7.5.6 Building Design Basis

It is interesting to note the types of facilities that use the pressure-venting concept. The ORR is a pool-type reactor that uses ordinary water as a coolant. This water circulates at a relatively low temperature (~125°F), so there is very little stored energy in it. Even if the entire coolant inventory were discharged to the building, a negligible pressure rise would occur (Table 7.47). This being true, an accident would be less severe than those previously analyzed, so less-stringent containment requirements are imposed. Since there is virtually no pressure rise, the fission products can be handled directly. It is assumed that the fission products would be released and dispersed through the building. Then, since the building is kept at a negative pressure, the building contents would move through a cleanup system and up the stack.

The system did have some weak points, but these have been modified. For instance, early in the operation of the ORR the building was not kept at negative pressure constantly, but the negative pressure was to be supplied by opening a damper to the stack blower system. However, the building is now constantly connected to the 6000-cfm vent duct, which keeps the building at a negative pressure. If this draft is lost, the reactor is shut down.

A second example of improvement is the scrubber in the cleanup system. Initially, the recirculation pump in the scrubber had to start if an accident occurred. The pump was tested periodically and found to start on demand each time it was tested. It was felt that the pressure drop of

Table 7.47. Moderator and Coolant Properties

| Reactor | Moderator | Coolant | Coolant Properties | | |
|---|---|---|---|---|---|
| | | | Quantity (lb) | Operating Conditions | |
| | | | | Pressure (psig) | Temperature (°F) |
| Hallam | Graphite | Sodium | | 58 (pump head) | 945 |
| ORR | Water | Water | 584,000 | Atmospheric | 125 (av) |
| Humboldt Bay | See Table 7.34 | | | | |
| Bodega Bay | See Table 7.34 | | | | |

the entire system needed to be reduced, so the scrubbers (which were required to start on demand) were replaced by charcoal filters. The installation of these charcoal filters also enhanced the iodine-removal efficiency of the cleanup system. Furthermore, the system will not depend upon anything "starting" for its proper operation, and this is an improvement.

The Hallam building was provided primarily to contain the refueling operation. The reactor is sodium cooled and graphite moderated. At operating conditions the temperature is high (reactor outlet 945°F), but the pressure is quite low. The reactor system is contained in interconnected rectangular cells underground with concrete walls lined with steel. These cells are nominally leaktight and have sufficient strength to withstand the greatest operating accident. The maximum operating accident is complete and instantaneous loss of primary coolant (160,000 lb of sodium) at a temperature of 945°F. The cells are filled with a nitrogen atmosphere, so no sodium fire is considered credible. The addition of heat to the system will increase the pressure in the cells to 3.9 psig, and the temperature will rise to 285°F. The cells are designed for a pressure of 6.0 psig. Since the likelihood of a severe fission-product release during operation of this type of reactor is small, the important thing to contain is the accident that might occur during refueling (see also Table 3.2).

The Humboldt Bay refueling building provides containment during the refueling operation, but it also provides control of leakage from the reactor containment structure in the event the pressure-suppression system is called upon to operate. As was discussed in Section 7.4, the reactor is contained by the pressure-suppression system, but the refueling operation cannot be carried out within that system because of space limitations and because the reactor must be shut down and the system open for refueling. In view of these considerations, a negative-pressure building containment was provided. The design bases for the various negative-pressure buildings are given in Table 7.48.

Table 7.48. Bases for Building Ventilation Design

| Facility | Design Bases[a] |
|---|---|
| Hallam refueling building | −1/8 in. $H_2O$ average building pressure; this requires a fan capacity of 3120 cfm for inleakage of 450% of the contained volume per day |
| ORR | −0.3 in. $H_2O$ average building pressure; this requires a fan capacity of 6000 cfm for inleakage of 1000% of the contained volume per day |
| Humboldt Bay refueling building | 1/4 in. $H_2O$ average building pressure; inleakage of 100% of the contained volume per day is assumed |
| Bodega Bay refueling building | 1/4 in. $H_2O$ average building pressure; inleakage of 100% of the contained volume per day is assumed |

[a]In the event of a very strong wind the external pressure on the downwind side of the building may be lower than the building pressure and thus cause exfiltration of the building contents. It is presumed that these conditions would result in dispersion of fission products but that the site boundary dose would not be exceeded.

### 7.5.7 The Maximum Accident

For both the Hallam and the Humboldt Bay refueling operations, the maximum accident is a dropped fuel element (see Table 7.49). The fuel-handling device is capable of carrying only one fuel element at a time, so it is not credible that more than one element would be involved in an accident. The release of fission products from the element depends upon heating and melting of the element cladding by the heat source within the irradiated element. It is reasonable to assume that something less than all the fission products contained in the hottest fuel element would be released to the building. At Humboldt Bay another possibility is considered, that of the fuel transfer cask falling into the pressure vessel and doing damage to the core.[52] If this occurred, about 25% of the core would be damaged, but the temperature in the elements would remain low; therefore only gaseous activity would be released. The possible releases from the stack are listed in Table 7.50. After cleanup by the filter system and the enhancement of dispersion by exhausting up the high stack, the doses calculated for the worst cases would be those listed in Tables 7.51 and 7.52.

In a condition of high wind velocity, a low pressure might be produced on the leeward side of the building that might be sufficient to cause leakage from the building at or near ground level; i.e., the low pressure on the lee side might be lower than the building pressure, and air leakage would occur. This occurrence was foreseen for the Humboldt Bay refueling building,[28] as indicated in Table 7.48.

In the unlikely event of the fuel-handling accident occurring coincident with a high wind velocity condition, the maximum lifetime thyroid dose calculated assuming 100% volume per day exfiltration from the refueling building at ground level (40-mph unstable wind) is approximately

Table 7.49. Fission-Product Release in the Event of an Accident

| Facility | Nature of Accident | Operating History | Release to Container | Release to Atmosphere |
|---|---|---|---|---|
| Hallam refueling building | One dropped fuel element and subsequent oxidation of that element | 1 yr at 1.5 Mw(t) with 10-hr cooldown before accident | $1.63 \times 10^5$ curies noble gases after 1 hr<br>$1.85 \times 10^5$ curies iodine after 1 hr | |
| ORR | Core meltdown | 39 days at 20 Mw(t) | 100% noble gases<br>100% halogens | No significant release from building |
| Humboldt Bay refueling building | One dropped fuel rod | Long term at 230 Mw(t) with shutdown of 8 hr prior to accident | 10% noble gases<br>2.5% halogens<br>1.5% volatile solids<br>0.03% other solids<br>(of the single rod inventory) | See Table 7.50 |
| Bodega Bay refueling building | Insertion of reactivity such that fuel melting occurs | Long-term fission products | | See Fig. 7.43 |

Table 7.50. Fission-Product-Release Rate from Humboldt Bay Stack Following Postulated Fuel-Handling Accident

| Time After Event | Release Rate (μc/sec) | | | |
|---|---|---|---|---|
| | Noble Gases | Halogens | Volatile Solids | Other Solids |
| 20 min | 0 | 0 | 0 | 0 |
| 1 hr | 180,000 | 2500 | 1500 | 200 |
| 3 hr | 160,000 | 2100 | 900 | 170 |
| 10 hr | 100,000 | 1300 | 600 | 120 |
| 1 day | 50,000 | 600 | 300 | 60 |
| 2 days | 13,000 | 130 | 80 | 20 |
| 3 days | 4,000 | 40 | 30 | 6 |

Table 7.51. Doses After Accident

| Facility | Direct | Submersion | Thyroid | Bone | Comments |
|---|---|---|---|---|---|
| Hallam refueling building | | | | | See Table 3.2 |
| ORR | | | | | See discussion |
| Humboldt Bay refueling building | See Table 7.52 | | 0.05 rem at 0.6 mile under unstable weather conditions | | These releases consider discharge at 250 ft above ground level and exposure time of 10 hr |
| | | | 0.5 rem at 5 miles under stable conditions | | |
| Bodega Bay refueling building | | 0.034 rem under unstable weather conditions | 0.04 millirem | | |
| | | 0.16 rem under stable conditions | 0.2 millirem | | |

Table 7.52. Direct Radiation from Humboldt Bay Building Following Fuel-Handling Accident

| Time After Event | Curies in Building | Dose Rate at 1/4 Mile (mrem/hr) |
|---|---|---|
| 1 hr | 22,000 | 0.5 |
| 3 hr | 19,000 | 0.4 |
| 10 hr | 12,000 | 0.3 |
| 1 day | 5,800 | 0.1 |
| 2 days | 1,500 | 0.04 |
| 3 days | 500 | 0.01 |

25 rems at the nearest site boundary for exposure during the first 10 hr. Wind variation and direction diversity would be expected to reduce this dose by a factor of 10.

The ORR Reactor and its containment building "paved" the way for the negative-pressure concept, and because of this, it has been in a continual state of reevaluation. It may be noted that few entries have been made in Tables 7.49 and 7.51 for the ORR, whose mca is discussed below.

In the ORR analysis, the dose to the thyroid as a function of distance was calculated based on the weather condition that would give the highest dose for a given distance. These values were plotted, as shown in Fig. 7.47. It should be pointed out that this curve gives the worst possible dose at any given point for a variety of weather conditions. Also, for this plot it was considered that all the iodine in the fuel escaped from the stack (250 ft) at a rate and concentration determined by the building volume, the fission-product concentration in the building, and the stack flow rate. Further, this plot was normalized to 1-kw operation.

In the worst possible case, the receiver would be 700 m from the reactor, and the reactor would have been operating for a long time at 30 Mw. This would result in an integrated dose (over the entire accident period) of 39,000 rems, if no credit were taken for any iodine removal. However, if it is conservatively assumed that 50% of the iodine escaped from the core and that the decontamination factor of the vent filter was 100 (it has been measured at greater than 100), the total integrated dose (TID) would be 195 rems for the entire accident.

For Bodega Bay, an accident is postulated that occurs during refueling operations. It consists of a number of coincident, independent refueling errors, including the dropping of a fuel assembly of maximum reactivity worth into a near-critical zone. This initiates an incident that releases fission products (see Table 7.49 and Fig. 7.43).

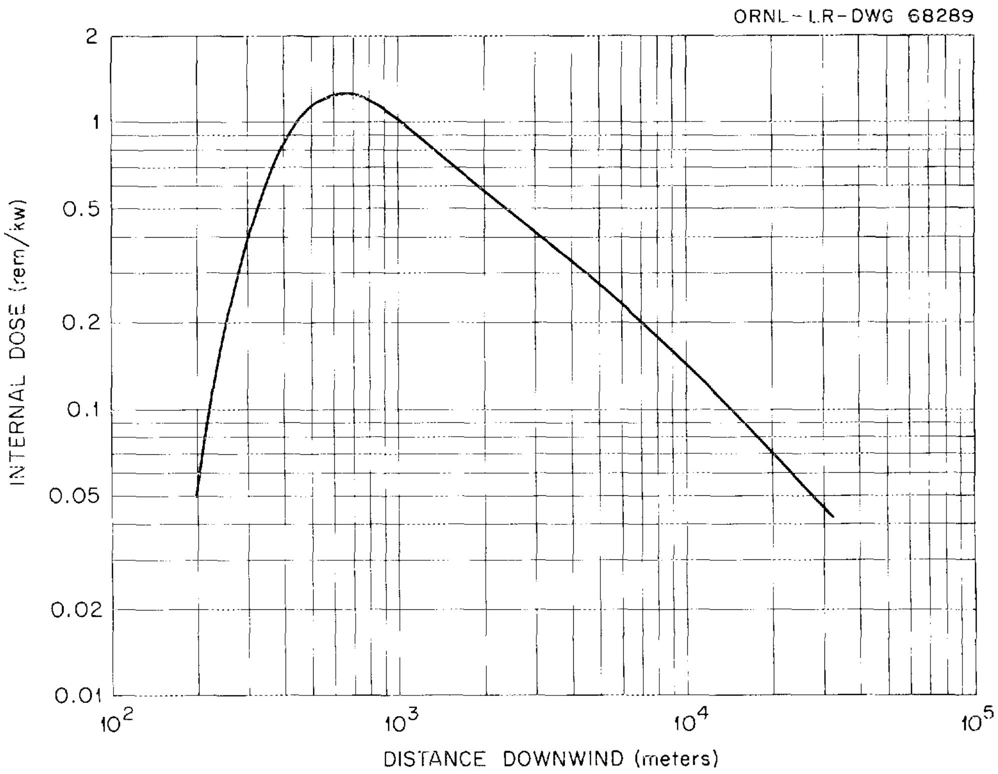



Fig. 7.47. Maximum Internal Dose from Iodine as a Function of Distance Downwind of the ORR. (From ref. 50)

There is the possibility of exfiltration of the Bodega Bay refueling building by a high wind:[47]

> "Assuming 100% volume per day exfiltration at ground level during a 40 mph unstable wind and assuming a 50% plate out factor in the leakage path for fission products other than noble gases, maximum halogen release rate to the atmosphere for the MCOA would be approximately 0.01 curies per second which could cause a maximum dose rate of approximately 0.01 rem to the thyroid per hour of exposure at one-half mile, the nearest point off-site which might be continuously occupied.
>
> "The direct radiation from the refueling building has been estimated based on preliminary design. For a refueling building inventory of 400,000 curies the estimated dose rate is 0.0004 rem/hr at 1/2 mile. The maximum continuous occupancy dose due to direct radiation would be approximately 0.01 rem at 1/2 mile."

At the EGCR, a type of pressure-venting containment may be used to minimize the effect of the postulated mca for that reactor, but the proposed scheme has not yet been approved by the Commission's regulatory staff. Upon occurrence of the mca the container would be isolated and the pressure caused by the accident energy would be withstood by the container. The peak pressure would rapidly abate because of condensation of vapor and conduction of heat through the container wall. Within the first hour, a nitrogen purge of 1000 scfm would be initiated to reduce the hazard of graphite oxidation. This purge would continue for 1 1/2 hr, at which time the rate would be reduced to 200 scfm; purging at this rate would continue for approximately 72 hr. Since the purge gas would be added to the isolated container, the container pressure would be increased. About 24 hr after the accident, the pressure would rise to 7.6 psig; at which time the contents of the container would be vented through a filter system and up the stack to the atmosphere. The capacity of this system is 750 scfm when the driving pressure is 7.6 psig, and the flow capacity decreases with a decreasing internal pressure. After 81 hr the container pressure would be 0 psig, and no further venting would be necessary.

## 7.6 PRESSURE-RELIEF CONTAINMENT

The pressure-relief containment system consists of two parts: (1) a system of venting ducts and (2) a building of low leakage. The venting ducts are large and can allow rapid transport of a large mass of gas with a small differential pressure. The building is similar to the high- or low-pressure containers (see Secs. 7.2 and 7.3) in that it is built to retain leaktightness up to possibly 10 psig. The maximum accident might release enough energy to cause a very high pressure, if the entire effect were to be contained, but the object of this system is to safely reduce the peak accident pressure. This is done by releasing the initial pressure peak to the atmosphere. The building is then isolated so that the remaining portion of the effects of the accident are contained in the more usual fashion. This system can be employed only when the pressure peak and major activity release are separated in time. This is, of course, not true for all reactor systems, and accordingly this containment concept cannot be used for all reactor types.

In analyzing the accident that dictates the use of a containment system, the following sequence of events must be considered. First, there would be a rupture of a primary coolant line that would allow rapid emptying of the coolant (water, in this case) and steam from the thermally hot reactor. Second, upon loss of the coolant, the fuel elements might melt due to their own decay and sensible heat. This would take a finite length of time. Third, the fission products would be released from the melted fuel elements, and they would escape from the primary system and diffuse through the building. This also would take a finite period of time. The principle involved in this containment system is to take credit for and make use of these finite periods of time to release the pressure before large amounts of fission products are made available for leakage from the building.

The steam to be vented may be contaminated by fission products that leaked from failed fuel elements or by activity from neutron activation, but this would represent a negligible hazard. If a serious amount of activity had built up in the coolant, the reactor would have been shut down. Since this concept allows release of the energy to the atmosphere, by the time the fission products escape from the core, the accompanying pressure would be reduced significantly. After this venting, but before the fission products escaped, the building would be closed by closing the vent pipes. In general, these systems are designed for about 5 psig in the isolated condition. To eliminate long-term pressure buildup by the decay heat source and possibly other sources, a spray system is available. This may be activated by pressure-control systems designed to maintain the building pressure between some limits; for example, the control system might activate the spray at 3 psig and turn it off when the pressure fell below 1 psig. This would tend to protect the building from overpressure, as well as pressure below atmospheric.

In many instances there are specific tasks that must be accomplished by an operator from the control room after an accident. Therefore the control room must be designed to remain tenable after the accident.

### 7.6.1 Building

The building to house the reactor and to afford this type of containment must meet specifications similar to those for low-pressure containment, that is, the building and penetrations should be capable of withstanding an internal pressure. A typical example is shown in Fig. 7.48 (ref. 53). Other design criteria such as wind load, seismic loads, etc., are dictated by local building codes.

The vent ducts are among the most important features of the system. These ducts must be depended upon to vent a large amount of gas from the building in a short time at some low pressure drop. The New Production Reactor at Hanford, Washington, depends upon several vent ports (see Chap. 9) to relieve the initial pressure surge. The NPD reactor (Fig. 7.49, ref. 54) at Chalk River in Canada has one very large vent duct, 9 by 12 by 130 ft long, that is designed to keep the building pressure below the design pressure of 4 psig at maximum flow conditions.[55] Much research work was done on this system using scaled-down models[55,56] (see also Sec. 9.6.7).

### 7.6.2 Tests

Leakage requirements and testing should follow the same criteria as those set forth for the high-integrity building, that is, high- and low-pressure containment and dynamic tests of the ducts and vent system. Although the system is to vent initially, it is expected to be vaportight when isolated. After isolation, this system may be thought of as just another application of the low-leakage container.

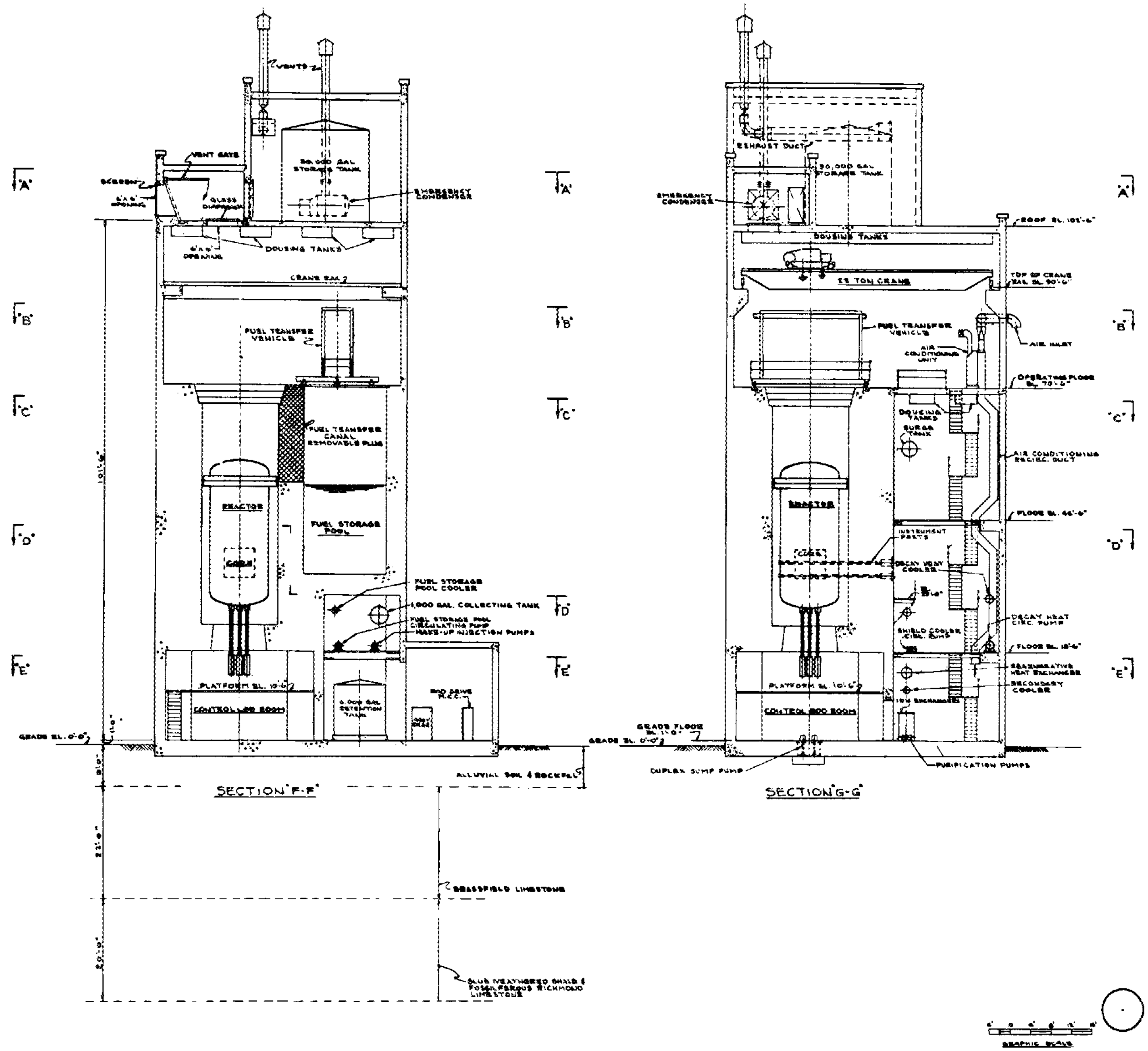



Fig. 7.48. Pressure-Relief Containment. (From ref. 53)

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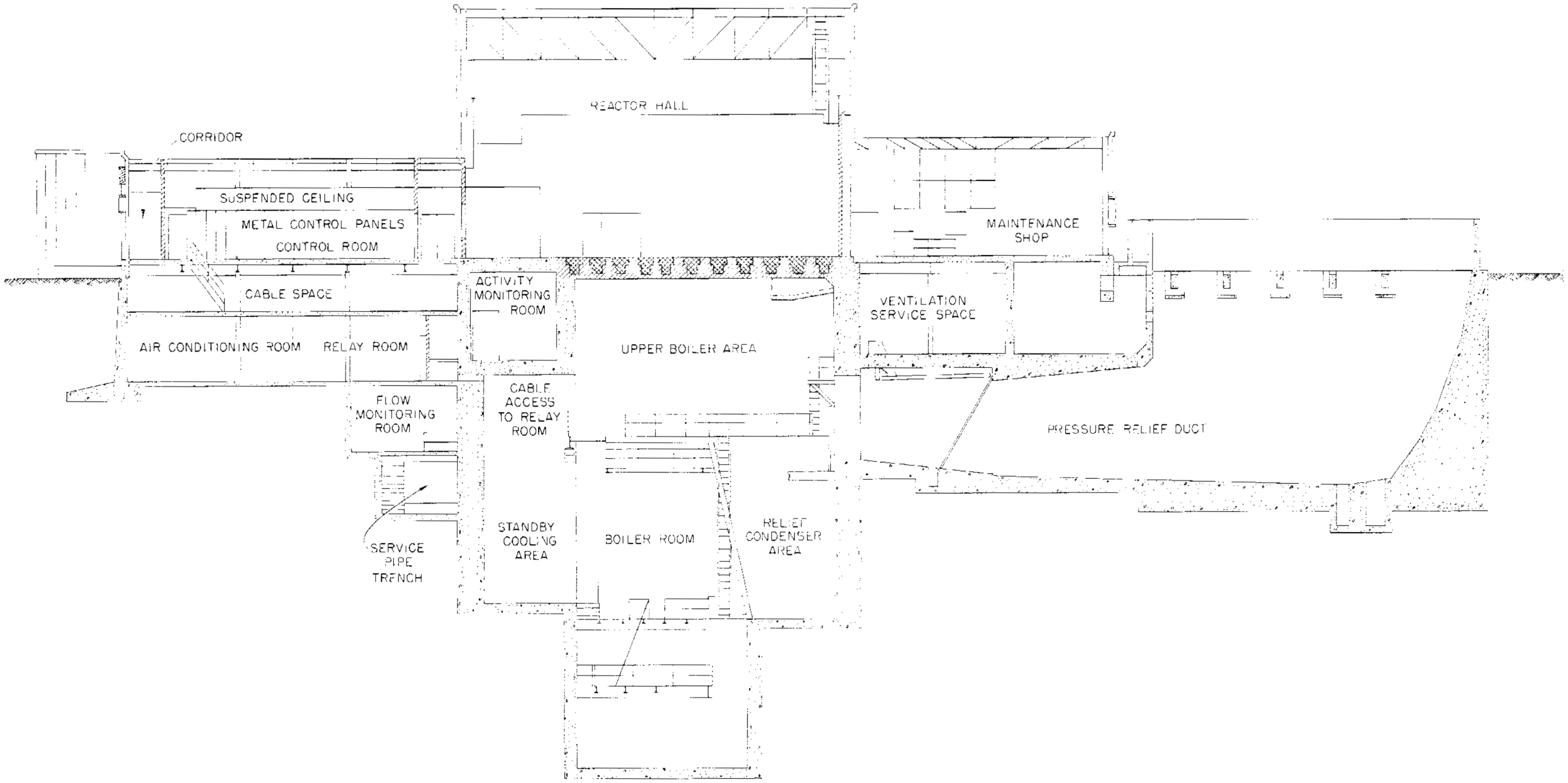



Fig. 7.49. The NPD Containment Building. (From ref. 54)

### 7.6.3 Penetrations

Air locks, doors, and other penetrations are similar to those discussed in Sections 7.3 and 7.4. Piping penetrations for the NPD will be bellows-sealed and electrical penetrations will be sealed in the conventional manner described in Chapter 9.

### 7.6.4 Accident Analysis

Since this is not an often used concept, little information is available at this time. The assumptions concerning the accident, which are generally made to determine design pressures, are similar to those made in previous sections, with the major difference being that the initial pressure peak for which high-pressure containers are designed will be vented. After venting and then resealing, the system will be expected to behave predictably with respect to leakage under the influence of the reduced pressure.

Under some unique conditions the vent ducts are opened to the atmosphere (see Sec. 9.6.7 for details), and the pressurized gases are released. Then the duct will be closed to effect isolation of the container so that the fission products that may be released later will be contained.

The closing of the valve may be initiated by one of possibly three signals. First, a calculation of the amount of time before fuel melting occurs may be made and this "delay" time used to close the valve. Second, a radiation monitor in the relief duct may cause vent closure. Third, the building pressure may be monitored, and when the pressure falls below some preset value, the vent will close. These signals, or a combination of them, could be depended upon for the desired action. In presently operating systems this action is not necessarily automatically initiated, and some emergency measures must be taken by the operators.

After closure of the ducts (i.e., building isolation is effected) and in case of core cooling system failure, the fuel may undergo the maximum melting and consequently release fission products to the building. The building spray system that would control the building pressure would, in the event of fission-product release, remove much of this activity from the building atmosphere by washdown. In any case the amount of activity available for release would be calculated using the usual conservative estimates for release, plateout, and washdown, and then this would be assumed to leak from the building. For the calculation of the site boundary doses, the ground-level release and the worst weather condition would be assumed.

## 7.7 UNDERGROUND CONTAINMENT

Underground containment per se is not necessarily a different containment type, since a reactor and its containment system could be located underground and still employ one of the three major containment types previously discussed. However, underground containment is discussed separately here because of the unique problems and features of such a location.

There are several considerations that make the placing of reactors underground appear attractive. The first is economics. In many European countries, where labor is relatively cheap and the cost of steel and building materials is high, it is now often economical to excavate a tunnel and place the reactor underground. In the United States it is presently more expensive to do this; but, as reactors are moved nearer to population centers, the costs of the site and necessary exclusion area will become quite high. This will offset some of the added cost of the underground container. Another factor is the relative invulnerability of underground plants to destruction by enemy attack. From the aesthetic point of view, the below-ground container would have a further advantage in that it would not appreciably change the appearance of the landscape.

### 7.7.1 Building Concept

This concept depends upon the geologic structure for strength to contain the maximum accident and to provide holdup time and partial cleanup of the fission-product vapors and gases. A suitable rock formation must be present.

The Halden boiling-water reactor in Norway,[57,58] the Lucens power plant in Switzerland,[59] and the Avesta reactor in Sweden are contained in rock caverns. In this country, a study[60] was made of the placement of the EBWR in an underground container.

The Halden container was excavated by conventional methods to the size and shape necessary for the installation. After this, repairs were made on the walls and ceiling. Weak rocks and cracks were tied to sturdier ones by means of anchor bolts and steel straps, voids were filled with grout, and other work such as this was carried out. Upon completion of this work, a concrete liner was poured; this added strength to the system and provided a smooth surface for application of paint or other lining material.

The shape of the cavern depends greatly upon the strength of the rock being excavated. For example, the Halden Reactor is located in rock with limited strength in the horizontal plane, so wide unsupported spans could not be used. This consideration dictated the shape, a long, rather narrow hall (Fig. 7.50, ref. 58).

### 7.7.2 Tests

Leakage-rate tests were performed at the Halden Reactor, mainly for the purposes of evaluating the permeability of the containment rock. This experiment was carried out at several pressures, and makeup air was injected through a calibrated nozzle to keep the pressure at the desired level. The volume of air added was calculated, and this was taken to be the leakage. At approximately 5 psig, a volumetric leakage rate of about 29% per day was indicated. However, most subterrainean caverns would require metal linings to meet U. S. leakage requirements.

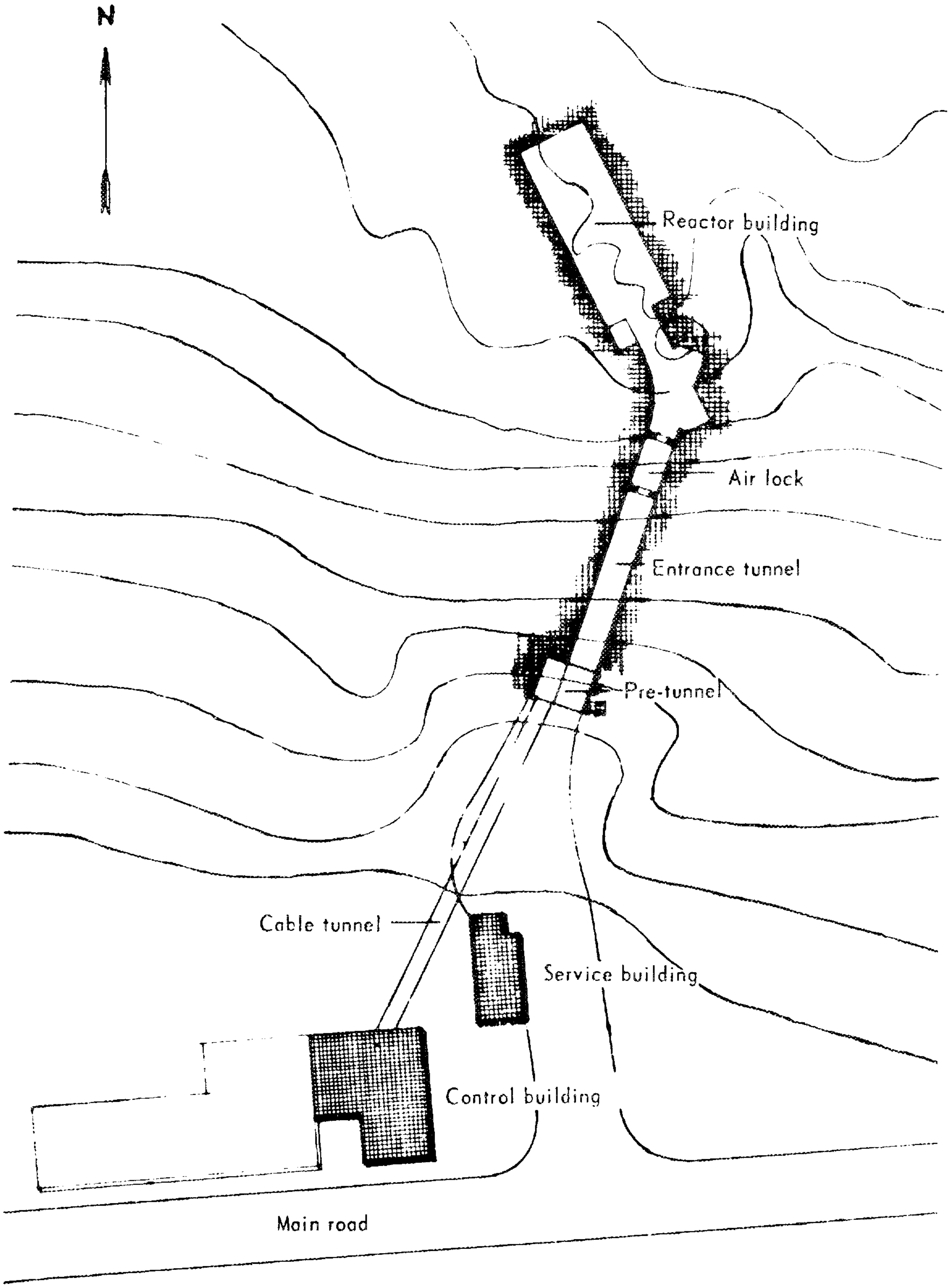



Fig. 7.50. Underground Reactor Building. (From ref. 58)

A study[61] was made of the leakage rates that might be expected from cavern containers in general. The conclusions of that study are:

> "...the efficacy of an underground enclosure of confinement rests on the following properties:
>
> -Its tightness is weak but adequate for ensuring that the leakages are slowed down over several hours;
>
> -The leakage of gases which flow through the concrete are stored in the pores of the rock over a very low thickness;
>
> -The progression of leakage gases follows a process of diffusion which is very slow;
>
> -The walls and the rock behave as a filter which only admits the passage of gases of fission and halogen vapors;
>
> -The gases of fission whose disintegration products could lead to a contamination of long duration are themselves short-lived;
>
> -The duration of storage of gases of fission and of halogen vapors in the rock is such that their radioactivity becomes negligible before they reach the external atmosphere.
>
> "Lastly, the underground placement of a nuclear power plant in a judiciously chosen mass of rock permits the confinement of the gravest accident by the utilization, solely, of procedures current in civil-engineering practice."

### 7.7.3 Penetrations

The only penetration into the reactor building is the entrance tunnel. This tunnel contains an air lock that is conventional in concept (i.e., double interlocking doors). Piping, electrical, and instrument penetrations are brought through the end walls of the air lock.

Special attention must be given to the installation of the bulkheads in the tunnel to form the opposite ends of the air lock. But since these bulkheads are steel, the piping and electrical penetrations are similar to those used for high-pressure containers (see Sec. 7.2).

### 7.7.4 Protection

The underground system is inherently safe from damage (to a catastrophic extent) by shock waves and internal missiles. One of the advantages of this concept is its almost complete invulnerability to damage by external missiles or explosions, such as one might expect during wartime.

### 7.7.5 Basis for Design

The accident that dictates the capabilities of the system is not affected by the type of container. The design accident for Halden is complete loss of coolant followed by adiabatic expansion of the coolant.

## 7.7.6 Doses

Doses have not been calculated for underground installations. It is clear that the direct dose would be insignificant because of the great amount of shielding afforded by the overburden. Leakage of gases from the system should not present a problem, since the escaping activity would be required to travel a tortuous path to reach the atmosphere, the holdup time would be great, and the probability for absorption and deposition would also be great.

Studies concerning the absorptivity of various types of earth[62,63] have been made using strontium and iodine as the products sorbed. Figures 7.51 and 7.52 (ref. 63) show the results of these tests for containment concentrations up to 900 mg/liter.

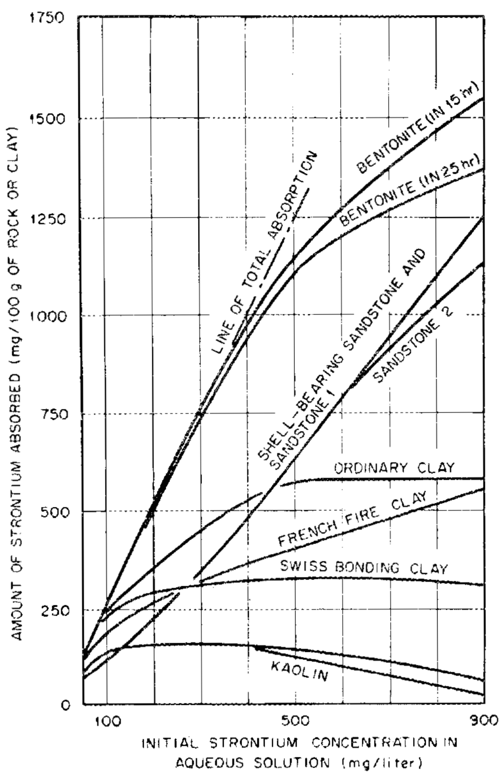



Fig. 7.51. Absorption of Strontium in Clays and Rocks. (From ref. 63)

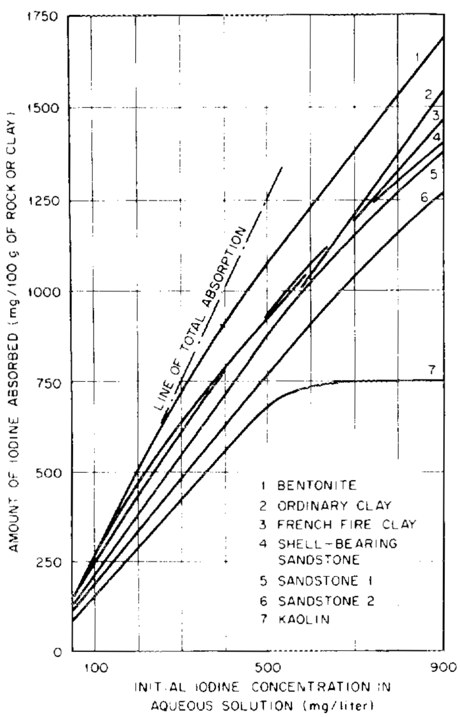



Fig. 7.52. Absorption of Iodine in Clays and Rocks. (From ref. 63)

## 7.8 MULTIPLE CONTAINMENT

Multiple containment denotes any one of a number of methods of providing two or more complete containment barriers around the primary reactor system. It may in the future be of significant economic advantage to build reactor-powered electrical generating units near population centers, and if this is to be done, a high degree of plant safety must be provided. In early 1963, such a plant was proposed that was to be built in the Borough of Queens in New York City, but the proposal has since been withdrawn because a cheaper power source became available. This plant, the Ravenswood Plant of Consolidated Edison, is described in Section 7.8.3. Plants that have provided a background of experience in the application of multiple containment are the NS Savannah and Indian Point (see Table 7.53). These plants were described in Section 7.2 because in both cases the inner container is of the high-pressure type. More recent proposals involving some forms of multiple containment include the Connecticut Yankee Reactor[64] and the City of Los Angeles Reactor.[65]

### 7.8.1 Indian Point

One of the first applications of the multiple containment concept was the Indian Point plant of the Consolidated Edison Company. This plant has a conventional high-pressure container (see Sec. 7.2), which is surrounded, or contained, by a concrete building. The primary purpose of the concrete building is to provide shielding, but it also provides a means of controlling the atmosphere in the space outside the inner container at a negative pressure. The atmosphere of this annular space can be exhausted up the stack. In the case of an accident the inner container

Table 7.53. Plants with Multiple Containment

| Reactor | Location | Thermal Power (Mw) | Type | Prime Contractor | Architect-Engineer | Containment Fabricator | Nuclear Equipment Supplier | Operator |
|---|---|---|---|---|---|---|---|---|
| NS Savannah | Mobile | 69 | PWR | Babcock & Wilcox | (a) | N.Y. Shipbuilding | Babcock & Wilcox | |
| Indian Point | Buchanan, N.Y. | 585 | PWR | (a) | (a) | Chicago Bridge and Iron | Westinghouse | Consolidated Edison |
| Ravenswood | Queens, N.Y. | 2030 | PWR | Westinghouse | (a) | (a) | Westinghouse | Consolidated Edison |

[a] Information not available.

may, within the specifications of the plant, leak fission products to the annular space. Some of the fission products will be deposited in the annular space, some particulates will be removed by the filter, and the remainder will be discharged at the height of the stack. These three mechanisms tend to lessen the dose to any given receiver. However, in the analysis of the mca for this facility, credit is taken only for holdup times provided by the annular space.

### 7.8.2 NS Savannah

The NS Savannah has the same sort of system, that is, a high-pressure container (see Sec. 7.2) which is located in a hold of the ship that is kept at a negative pressure. The primary advancement from the Indian Point system is in the cleanup of the air exhausted from the annular space. In addition to the absolute filters (as for Indian Point), charcoal filters for the removal of radioiodine are provided, as well as a more sophisticated system of blowers and ducts.

### 7.8.3 Ravenswood

The Ravenswood containment system goes a step further in the development of the multiple containment concept. The essential parts of the scheme are shown in Fig. 7.53 (ref. 66). The containment system consists of two welded-steel, low-leakage membranes separated by porous concrete, a "pump-back" system, and a system of fans, filters, and a stack. The pump-back system keeps the space between the two membranes at a negative pressure. Any leakage into this space is pumped back into the container. In this way, any fission products that leaked through the inner membrane would be returned to the container. If it were necessary to vent the container, this could be done by filtering, monitoring, and exhausting up the stack. However, the container would be exhausted to the atmosphere only if the concentration of fission products was below that permissible for release to unrestricted areas. Otherwise the container atmosphere would be transferred into special mobile containers so that it could be transported for disposal at a remote location.[66] However, the City of Los Angeles Plant [~1400 Mw(t)] is identical in concept and is now under consideration by the AEC for construction at a site within 30 miles of Los Angeles.

### 7.8.4 Design Parameters

#### 7.8.4.1 Building Design

The criteria of construction of the NS Savannah and Indian Point containers were discussed in Section 7.2. The Ravenswood plant is designed for a 40-psig internal pressure in the event of the maximum accident. The entire force of this internal pressure is to be resisted by the 5 1/2 ft of reinforced concrete that completely surrounds the two steel membranes and the porous concrete. The two membranes provide

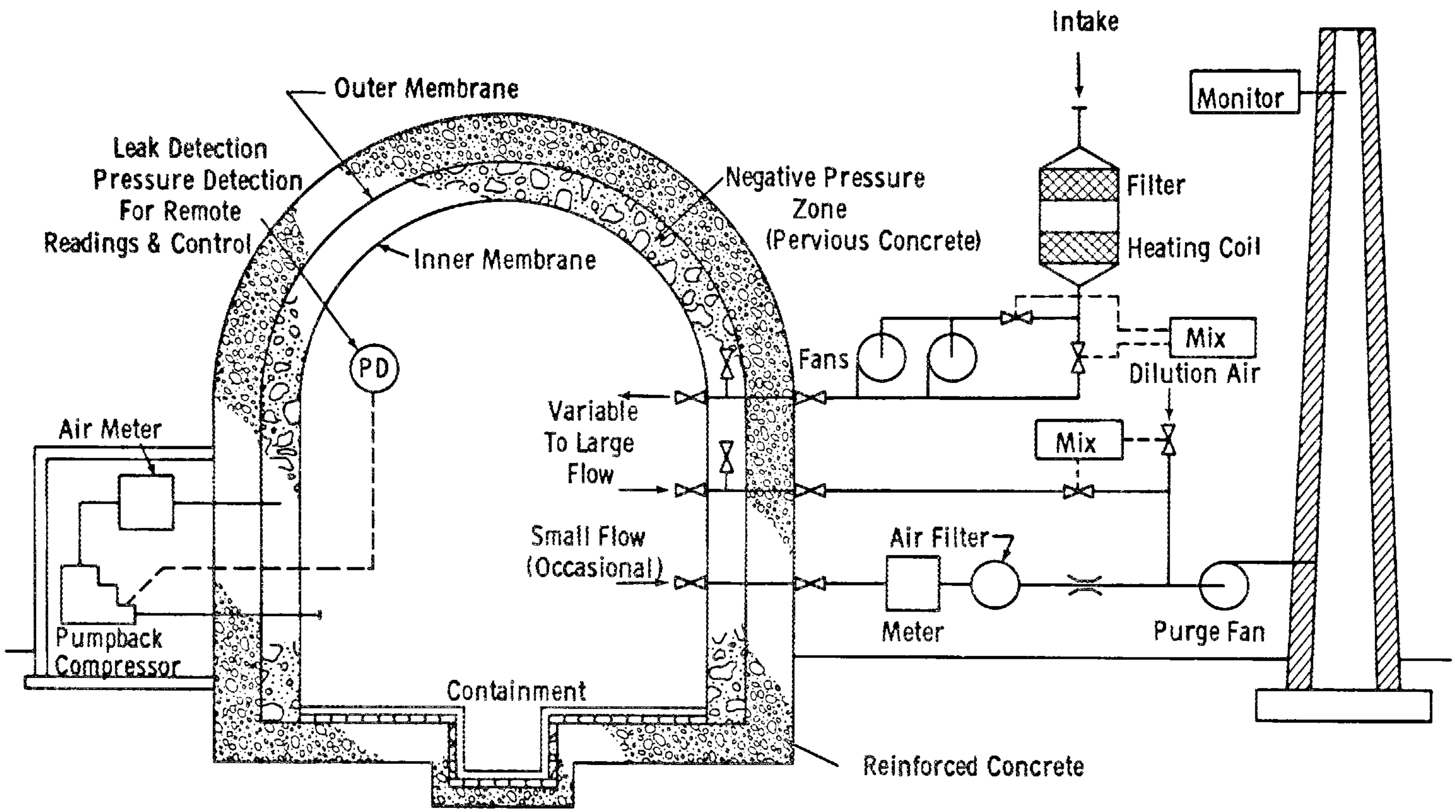



Fig. 7.53. Diagram of Reactor Containment System for the Ravenswood Plant. (From ref. 66)

leaktightness, and the reinforced concrete provides strength (see Fig. 7.54, ref. 63). The free volume and therefore the physical size of the containment structure were determined by assuming the accidental and instantaneous release of all the primary coolant into the container and allowing the pressure not to exceed 40 psig. Physical loads from other sources (i.e., wind load, seismic load, etc.) would be small by comparison but were also considered. This information is presented in Table 7.54.

#### 7.8.4.2 Vacuum System

The Indian Point and NS Savannah plants are provided with exhausting fans for the spaces surrounding the container. The NS Savannah recently tested the fan system and found that a 3-in.-$H_2O$ vacuum could be maintained with both fans operating.

The pump-back system of the Ravenswood plant was designed so that any one of three vacuum pumps could deliver the desired 10-in.-Hg vacuum with a flow rate of 8 cfm. It must be noted that pump-back systems such as these pump back not only leakage from the inner container but also that from the atmosphere into the outer annulus. The latter volume, however, is limited by the pressure differential. The containment purge system would be used to purge the containment atmosphere after operation and before entry by personnel. It could also be used to clean up the containment atmosphere in the event of a minor release. The system consists of ducts, filters, and two 40,000-scfm fans.

### 7.8.5 Proof Tests

The tests performed on the Indian Point and NS Savannah containers were discussed in Section 7.2. An integrated leakage-rate test at 15 psig was planned for on the finished container of Ravenswood. Tests of individual penetrations would also be performed periodically. These tests are described in Table 7.55.

### 7.8.6 Material Specifications

Specifications and the materials of construction are given in Table 7.56 for Indian Point, NS Savannah, and Ravenswood. The reinforced concrete of the pressure-restraining portion of the Ravenswood container would be prepared and installed according to the standards of the American Concrete Institute (see Chap. 2).

### 7.8.7 Penetrations

Information concerning the number and type of penetrations is not available for the Ravenswood plant. This information for the NS Savannah and Indian Point reactors may be found in Section 7.2.6. Penetrations for the Ravenswood plant differ from most high-pressure penetrations only in that two leaktight membranes are penetrated. Figure 7.55 (ref. 66) shows typical penetrations. It may be seen from this figure that the penetration is vented to the low-pressure space between the two membranes.

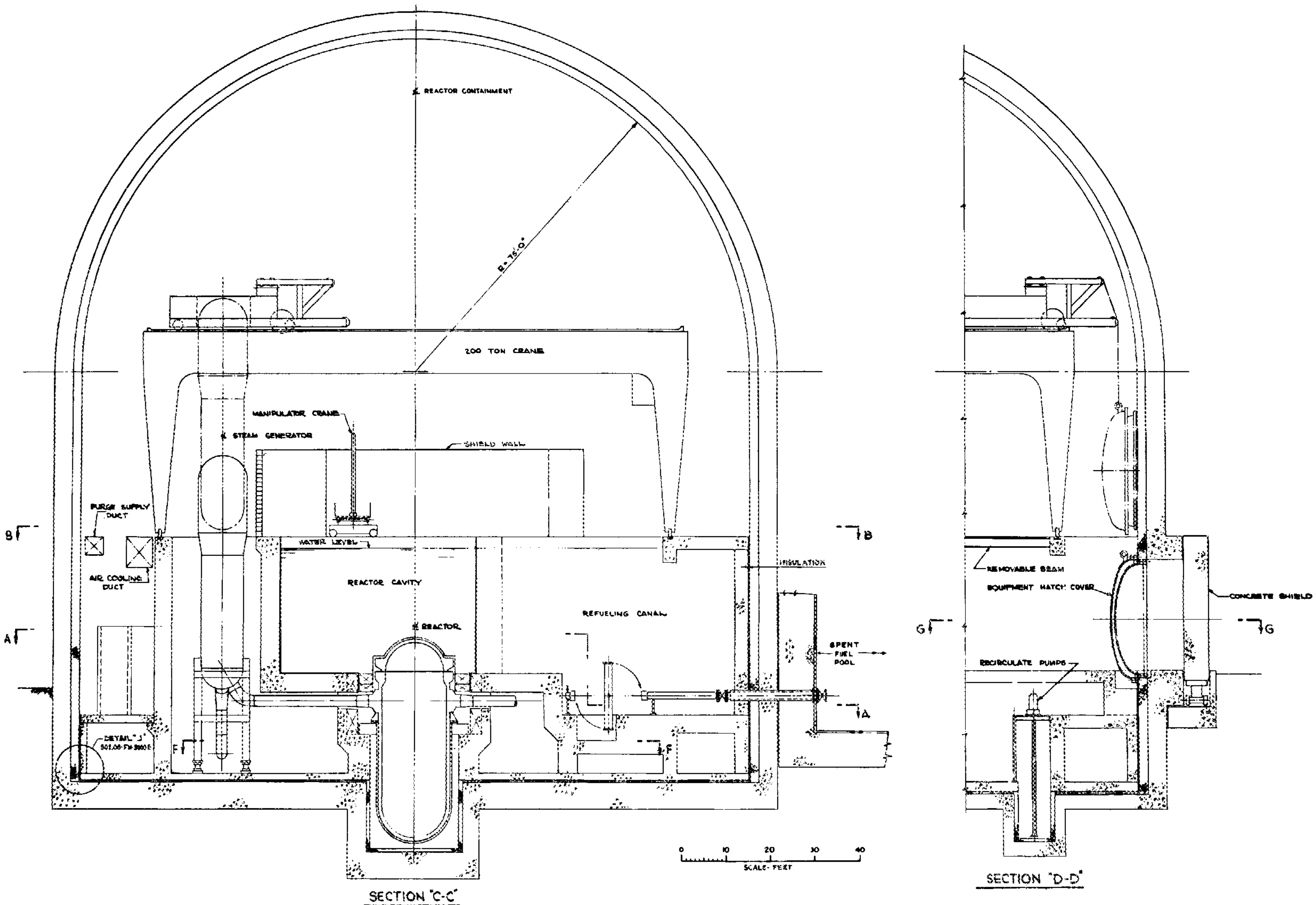



Fig. 7.54. Ravenswood Containment Building. (From ref. 66)

Table 7.54. Design Parameters for Multiple Containment

| Reactor | Shape | Free Volume ($ft^3$) | Dimensions | Design Pressure (psig) | Temperature (°F) | Vacuum Breakers | Wind Load | Seismic Load |
|---|---|---|---|---|---|---|---|---|
| | | $\times 10^3$ | | | | | | |
| NS Savannah | Reactor vessel: horizontal cylinder | 32.3 | 35 ft in diameter, 50 ft long | 186 (~100 ft of sea water) | 360 | | (a) | (a) |
| | Outer container: a ship's hold that is rectangular in shape and of usual ship-type construction | | | | | | | |
| Indian Point | Reactor vessel: a steel sphere partially below grade<br>Outer container: a concrete cylinder with an arched concrete roof | 2140 | 160 ft in diameter | 27.5 | 220 | Yes | (a) | |
| Ravenswood | Both inner and outer containers are welded steel of low leakage; containers separated by ~2 ft of porous concrete | 2580 | ~150 ft ID, ~167 ft high | 40 | 280 | | Hurricane wind velocities | |

[a] Does not apply.

Table 7.55. Container Leakage Tests

| Reactor | Container Tested | Leakage Test |
|---|---|---|
| NS Savannah | Outer | With the installed fans, the maximum vacuum obtained in the reactor compartment was −3 in. $H_2O$ with two fans and −0.7 in. $H_2O$ with one fan operating |
| | Inner, see Table 7.3 | |
| Indian Point | Outer | Since the outer container would be used only as a holdup volume, no leakage data are available and no leakage rate is specified |
| | Inner, see Table 7.3 | |
| Ravenswood | Outer and inner | Both steel membranes will be leak tested at 15 psig; the permissible leakage rate is <0.1% of the contained volumes in 24 hr |

Table 7.56. Vessel Material Specifications

| Reactor | Container Material | | | Weld Inspection | Reactor Pressure Vessel | | |
|---|---|---|---|---|---|---|---|
| | Code | Specification | Thickness | | Inside Diameter (in.) | Wall Thickness (in.) | Material |
| NS Savannah | Inner: ASME | | | | 98 | 6 1/2 | A-212-B |
| | Outer: ordinary ship construction | | | | | | |
| Indian Point | Inner: ASME | A-201 A300 | 0.89 in. (upper) | 100% x-rayed | 117 | 6.94 | A-212B |
| | | | 1.03 in. (lower) | | | | |
| Ravenswood | ACI-318 | | 5 1/2 ft | (a) | | | |

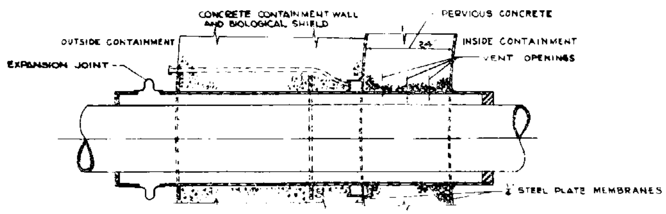
[a]Does not apply.

This means that any leakage of that penetration either from the container atmosphere or from the outside would be pumped back to the container. Provisions will be made to test these penetrations individually.

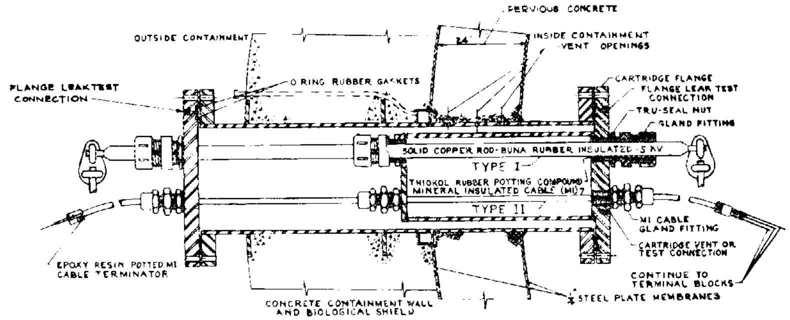
### 7.8.8 Building Protection

The measures taken to protect the building from various damaging mechanisms are listed in Table 7.57. The protection techniques used for the NS Savannah and Indian Point are discussed in Section 7.2.7. The Ravenswood plant is protected from exterior damage by the 5 1/2-ft-thick reinforced-concrete shell.

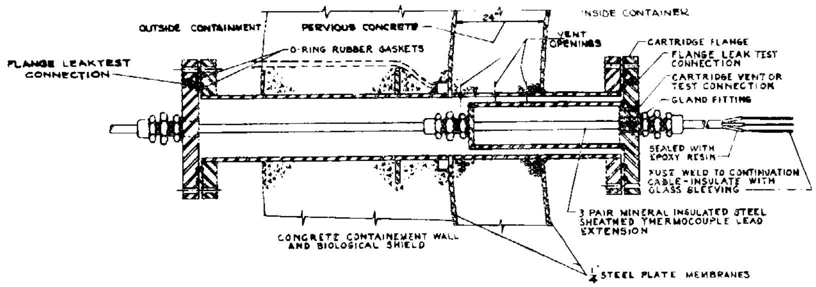
The possibility of a missile's being generated from some internal source is extremely remote. Potential missiles are, however, anchored



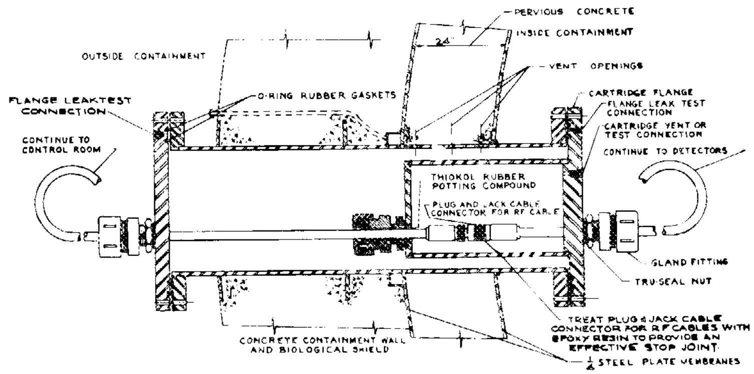

PIPE PENETRATION




ELECTRICAL PENETRATION
TYPES I AND II
2300 & 600 V CONVENTIONAL CIRCUITS




ELECTRICAL PENETRATION
TYPE III
THERMOCOUPLE LEAD EXTENSION CIRCUITS




ELECTRICAL PENETRATION
TYPE IV
COAXIAL & TRIAXIAL CABLE CIRCUITS

SCALE - NONE

Fig. 7.55. Penetrations for the Ravenswood Plant. (From ref. 66)

Table 7.57. Containment Structure External and Missile Protection

| Reactor | External Protection | Missile Protection |
|---|---|---|
| NS Savannah | Housed in a ship's hold | |
| Indian Point | Exterior is concrete | |
| Ravenswood | Exterior is concrete and needs no added protection | Internal: by tying down or supplying shields for potential missiles and by concrete partitions, etc. External: the 5 1/2 ft of reinforced concrete will protect the membrane from any credible missile |

in some manner or suitable missile shields are provided. The possibility of a missile of external origin has, likewise, been considered.[66]

> "The possibility of aircraft collision with the reactor containment has been considered. No direct hit by any presently known civilian or military aircraft can penetrate the outer 66 in. thick reinforced concrete shell.
>
> "Calculations have been performed to show that a 1500 lb turbojet rotor from an aircraft traveling at 150 per cent of the speed of sound will penetrate only to the depth of between 30 and 56 inches. This is based on the assumption that the rotor remains intact, a condition not considered credible. The calculations are based on formulas developed by the U.S. Navy and the Ballistics Research Laboratory, Aberdeen, Maryland."

### 7.8.9 Moderator and Coolant

The average thermodynamic properties of the primary coolant, which is assumed to be lost at the time of the accident, are listed in Table 7.58.

### 7.8.10 Design Accident

The design accident for the Ravenswood plant is similar to that of other pressurized-water reactors (see Table 7.59). It consists of instantaneous and adiabatic release of the primary coolant to the free volume of the container. This would produce a pressure peak of 40 psig. The core-injection system is assumed not to operate, so core meltdown would occur. The peak pressure would be reduced by the internal spray system.

Table 7.58. Moderator and Coolant at Assumed Accident Conditions

| Reactor | Type | Moderator | Coolant | Coolant Properties at Assumed Accident Conditions | | |
|---|---|---|---|---|---|---|
| | | | | Quantity (lb) | Pressure (psia) | Temperature (°F) |
| NS Savannah | PWR | | | | | |
| Primary | | $H_2O$ | $H_2O$ | 66,000 | 1750 | 508 |
| Secondary | | | $H_2O$ | 8,000 | 485 | 463 |
| Indian Point | PWR | $H_2O$ | $H_2O$ | 163,000 | 1500 | 500 |
| Ravenswood | PWR | $H_2O$ | $H_2O$ | $56.6 \times 10^4$ | 2035 | 570 (av) |

Table 7.59. Maximum Accident

| Reactor | Description | Energy Sources (Btu) | | | Core Injection | | Building Spray | | Comments |
|---|---|---|---|---|---|---|---|---|---|
| | | Chemical | Nuclear | Stored | Installed | Used | Installed | Used | |
| | | | $\times 10^6$ | $\times 10^6$ | | | | | |
| NS Savannah | Release of primary coolant through largest pipe (12 9/16 in. ID) plus the contents of one steam generator | 0 | 0 | 37.9 | No | | No | | The net result of release of primary and secondary contents is shown in Fig. 7.32 |
| Indian Point | Instantaneous release of all primary fluid and secondary fluid from one boiler | 0 | 43 | 130.5 | No | | Yes | No | The building spray system is assumed not to operate in the postulated accident |
| Ravenswood | Entire contents of primary system escapes instantaneously to the containment | 0 | 0 | 325 | Yes | No | Yes | Yes | The core injection system fails and the core melts completely; the building spray operates to remove fission products by scrubbing |

This system consists of a sump in the floor of the containment building, a heat exchanger to cool the spray water, a pump to supply sufficient head, and associated piping. The spilled primary coolant and any emergency-injection-system water that found its way into the container would be collected in the sump. This water would be cooled and pumped to the spray system and subsequently recycled until the spray system was shut down. By operation of the spray system, the effects of the maximum accident would be greatly reduced, but the double membrane and the pump-back system are designed to be sufficient to cope with the maximum accident without the aid of the spray system.

The description and discussion of the design accident for the NS Savannah and Indian Point reactors are given in Tables 7.9 and 7.59 and in Section 7.2.9.

## 7.8.11 Fission-Product Release

It was stated in the previous paragraph that the core injection did not operate after the loss of coolant, so the entire core was assumed to melt. The fission products that escaped from the melted core were taken to be 100% of the noble gases, 50% of the halogens, and 1% of the solid fission products (see Table 7.60; also example given in Part 100 of Title 10 of the Code of Federal Regulations). The Ravenswood containment system, on the other hand, was considered to function properly, and therefore no fission products would escape to the environs. Assuming this to be true, the only source of exposure would be direct radiation, and the 7 1/2 ft of concrete of the containment building walls would provide enough shielding to render this dose negligible (see Table 7.61).

### 7.8.11.1 Indian Point

The leakage from the Indian Point container would be held up by the concrete shielding building which surrounds the containment sphere. From this point the contents of the shielding building would be exhausted up the stack. By this holdup of the leakage from the container, the fission-product release to the environs would be greatly reduced. Furthermore, by releasing the effluent at the level of the high stack, the dispersion of released material would be greatly enhanced and the dose at the site boundary would be further reduced.

### 7.8.11.2 NS Savannah

The NS Savannah system is similar to that of Indian Point, but in addition to the mechanisms stated above (i.e., filtering and discharging up a stack), the Savannah also has charcoal filters to remove radioiodine from the effluent gases. A further advantage of the Savannah containment, which is unique to mobile plants, is that being aboard a ship, it can be moved to a remote location in case of an accident.

Table 7.60. Fission-Product Release in the Event of an Accident

| Reactor | Operating History | Fission-Product Release to Container (%) | Fission Products Available for Release from Container (%) | Remarks |
|---|---|---|---|---|
| NS Savannah | Dictated by port to be entered | 100 noble gases<br>50 halogens<br>1 solids | 100 noble gases<br>25 halogens<br>0.0005 solids | The assumed removal is by plateout; filter removal of 99% of the remaining iodine is also assumed |
| Indian Point | 600 days at 585 Mw(t) | 100 noble gases<br>100 halogens<br>10 others | 100 noble gases<br>100 halogens<br>10 others | By use of fans to evacuate the annular space (between the container and shielding building), the leaked fission products could be filtered and discharged up the stack |
| Ravenswood | 21 months at 2030 Mw(t) | 100 noble gases<br>50 halogens<br>1 others | 100 noble gases<br>25 halogens<br>0.5 others | 50% of the released halogens and others assumed to plateout or washout as in the example given in USAEC report TID-14844 |

Table 7.61. Doses from Fission-Product Release from Containment Structure

| Reactor | Direct | Submersion | Thyroid | Comments |
|---|---|---|---|---|
| NS Savannah | 1.8 rems/hr on the ship's deck (highest) | | | |
| Indian Point | | 3.4 r gamma<br>7.0 r beta at 1 hr | 32 r | |
| Ravenswood | Total integrated dose is 15 mr in 1 month at 50 ft | 0 | 0 | No fission products allowed to escape from the containment system; weather conditions would have no effect on the severity of the doses |

## 7.9 ENGINEERED SAFEGUARDS

In the event of a major reactor accident, core overheating might occur and result in the release of fission products to the reactor containment system. These fission products might then be distributed throughout the container and become available for leakage to the environment through the various leakage paths that may be present in the containment shell. Efforts toward greater reactor safety by improved mechanical devices or other engineered safeguards for preventing the release of fission products are encouraged by the AEC (see Part 100 of Title 10 of the Code of Federal Regulations). Some of the techniques are described in this section.

The prevention or reduction of fission-product release to the environment may be accomplished by three general methods: (1) preventing or minimizing, by emergency cooling or otherwise, the overheating of the fuel materials, (2) removing the fission products from the containment atmosphere by filtering, scrubbing, etc., and (3) constructing two or more barriers around the primary system so that the probability that a large quantity of fission-product activity may leak out is negligible or, at least, greatly reduced. The latter method was reviewed in Section 7.8. (A fourth method, that of scavenging radioactivity from the atmosphere after its release, has received some consideration,[67] but it has not proven practicable and therefore is not considered further here.)

Other engineered safeguards that may come to prominence in the future include means for limiting the size or seriousness of a rupture of the primary coolant system. These may take the form of minimized pipe size or ultraconservative design with respect to piping stresses and piping supports. An example of the former may be the pressure-tube type of reactor in which the primary coolant is introduced to the core through several small-diameter pipes. The rupture of one of these pipes would not allow a sudden and serious loss of coolant, and the rupture of many of these pipes is not considered to be credible.

The necessity of a high level of dependability in engineered safeguard systems cannot be overemphasized.[68] If credit is to be taken in the event of an mca, the dependability and durability of a given system must be demonstrated beforehand and be expected to be retained throughout its lifetime. Tests should be performed on a system to demonstrate conformance to specifications in the final installed condition and, as nearly as possible, under the accident conditions specified. Furthermore, the safeguards system must be applied to the whole reactor system, both under normal operating conditions and under accident conditions. This means that no mechanism, under any credible accident condition, should be able to bypass the safeguard system or reduce or eliminate its usefulness. After proper design, installation, and testing of the safeguard system are completed, continued surveillance, including monitoring, where applicable, must be exercised to assure the constant availability of the system. Finally, there must be assurance that some consequence of the maximum accident will not set in motion mechanisms that will destroy the safeguards system (e.g., a fire in a filter caused by decay heat from fission products it was designed to retain). These important points of safeguard dependability are outlined below:

I. Performance must be demonstrated

   a. Specifications must be met

   b. Performance of installed systems must be demonstrated

   c. Performance under simulated accident conditions must be demonstrated

II. Systems must be periodically shown to perform the desired function under any credible accident condition

   a. No "gaps" or weak points in the system coverage

   b. Must be protected from damage from all accidents, including the mca

   c. Must be designed to operate for the duration of the accident

III. Vigilance against deterioration and demonstration of continued availability must be provided

### 7.9.1 Core Cooling Systems

If a loss-of-coolant accident occurred, there would be depressurization of the reactor primary system, as well as severe reduction of heat transfer from the core. Heat generation due to fission-product decay would continue within the fuel elements, and the temperature in a high-power density reactor would rise. This temperature increase might be sufficient to melt the cladding of a number of fuel elements and make fission products available for release to the containment. In some cases it might be possible for the heat generation to cause volumetric expansion of encapsulated fission gases that would create an unusually high pressure inside the fuel elements. This could cause the fuel element to burst (see also Chap. 4). Regardless of the mechanism of failure, whether by melting or bursting, the chief interest is in limiting the temperature rise in the uncovered or uncooled core.

In order to eliminate or reduce the chance that a fuel element would fail, a system that will supply emergency cooling for the core is included in most reactors. The approach taken to this problem by several reactor plants is discussed in Table 7.62.

The emergency cooling system must be designed so that maximum reliability is attained in order for its function to be performed properly, since the only time that the system is called upon to operate is in the event of an accident that could result in a serious release of activity. If the initiating accident is able to void the emergency system, then it cannot be relied upon to mitigate the accident effects. For example,[68,69] the SL-1 accident included such severe core tank movement that all nozzles to the tank were severed, including those serving the core cooling system. If, in this case, the core cooling system had been needed to prevent core melting, it would not have been available.

The emergency core cooling system must be designed so that it and its associated instrumentation will operate reliably regardless of the credible

Table 7.62. Core Spray or Injection System

| Reactor | Addition or Circulation Rate | Spray or Injection Arrangement | Source of Coolant | Comments |
|---|---|---|---|---|
| Big Rock Point | 1000 gpm maximum addition rate<br>400 gpm recirculation rate | Spray ring in reactor vessel | Additional water is from Lake Michigan; supplied by electric or diesel fire pumps, each rated at 1000 gpm; after water reaches a certain level in the sphere, one of two 400-gpm recirculation pumps may be placed in service; these pumps take their suction from several locations throughout the bottom portion of the sphere | Cooling limits core melting |
| CVTR | 1500 gpm at 2185 ft of $D_2O$ | Injection into inlet headers (secondary injection into primary pump suction) | System recirculates water from the moderator tank and a sump in the bottom of the containment vessel | No credit taken in mca analysis |
| Elk River | 12 gpm minimum for 500 min | Spray ring in reactor vessel | 30,000-gal demineralized water storage tank | Core spray is depended upon to limit the extent of core melting |
| EGCR | 8000 lb of nitrogen per hr | Inlet and outlet nozzles in core vessel | An emergency cooling loop recirculates $N_2$ at a rate of 8000 lb/hr through the core; makeup provided by a purging system having a supply of 1 1/2 × $10^6$ scf of $N_2$ available; purge addition will normally be 200 scfm | Prevents fuel element failures and "runaway" air oxidation of graphite |
| PRTR | Two systems:<br>700 gpm at 400 psi | Equal division of flow to upper and lower ring headers and thence to reactor tubes | Reactor process water backed up by sanitary water supply, both from Columbia River; normal power supply | |
| | 500 gpm at 100 psi | Equal division of flow to upper and lower ring headers and thence to reactor tubes | Emergency water from well; diesel-driven pump | Hazards analysis based on this supply |
| Saxton | 375 gpm at 1450 psi (maximum) | Inject through inlet and outlet nozzles | From 80,000-gal storage tank | No credit taken in mca analysis |
| Yankee | 1800 gpm | Injection into each of the four main coolant loops | 117,000 gal of borated water is available, but it is considered the supply is unlimited as makeup is available | No credit taken in mca analysis |
| BONUS | 100 gpm (maximum) | 20 nozzles direct spray, 60% to fuel and 40% to superheater elements | Gravity fed (100 ft $H_2O$) from 100,000-gal storage which is shared by (1) core spray, (2) external building spray, (3) internal building spray, and (4) fire protection; additional makeup can be supplied from wells and pipeline | No credit taken in mca analysis |
| Humboldt Bay | 350 gpm at 140 psi | Spray ring near the top of the pressure vessel (see Fig. 7.56) | Spray water is taken from the suppression pool; there is more than enough water available to fill the reactor dry well to the vent openings, at which time the water will spill back into the suppression pool | No credit taken in mca analysis |
| NPD | 300 Igpm | Injection of makeup heavy water to primary system | Supplied by pumps from dump tanks to coolant inlet headers or pumped from the recovery well at the bottom of the boiler room | Credit taken for pressure reduction |
| | 2000 Igpm (max) | Injection of light water to the headers of the primary system | Supplied from 250,000 imperial gallons storage tank under gravity (150,000 gal for light-water injection; 100,000 gal for dousing) | |

accidents that may befall the reactor. Furthermore, reliability of operation should extend over the period of time that melting may occur.[70] In a water-moderated reactor in which the power density is large and the heat capacity of structure and moderation is not great, a large amount of cooling is necessary very quickly. The gas-cooled graphite-moderated system has different characteristics. The cooling is not needed immediately, but the cooling capability must be present over a much longer period of time for protection of the graphite. Thus, the reactor system demands specify the speed with which the cooling must be initiated and the length of time the system must operate reliably. The emergency core cooling system must initiate cooling within a given time and continue this cooling at least until any danger of meltdown or fuel element bursting has passed.

### 7.9.1.1 Water Reactor Systems

Emergency core cooling systems are used with the various water-cooled reactors. Water is sprayed or deluged into the core and allowed to flow downward past the hot fuel elements to cool their surfaces.

At the CVTR, water can be pumped into the reactor at two locations, the suction side of the primary circulation pump and the distribution header to the various pressure tubes. The water would flow down the pressure tubes and cool the fuel elements. It could then spill back to the container (through the break in the piping that caused the loss of coolant), where it would be collected in a sump for recirculation, or it could spill into the moderator tank, where a second recirculation system takes its suction.

The Humboldt Bay core spray system is arranged so that water would be injected directly onto the fuel elements by nozzles attached to a circular header near the top of the pressure vessel (see Fig. 7.56, ref. 28). The water would be supplied from the suppression pool, which is part of the accident containment system (see Sec. 7.4). When the reactor cavity was eventually filled up to the vent pipe entrances by the continued operation of the spray system, the water would spill back into the suppression pool through the dry-well vent pipes. There is sufficient water in the suppression pool to fill the reactor dry well up to the vent pipe entrances without uncovering the discharge ends of the vent pipes.

The BONUS core spray system consists of headers and nozzles (see Fig. 7.57, ref. 42) that would distribute the water flow to the various parts of the core. The superheater elements would receive 40% of the total, and the active core elements would receive the remainder.

### 7.9.1.2 Gas-Cooled Systems

Upon depressurization of the primary loop, with possible loss of coolant, there would be a severe reduction in core cooling that would cause the fuel elements and moderator of a gas-cooled reactor to heat up as the reactor afterheat began to build up. Emergency cooling might be required to ensure that abnormal fission product release was eliminated or reduced. In many cases, thermal convection through the core would afford sufficient cooling; but if this would not yield the desired results, an auxiliary

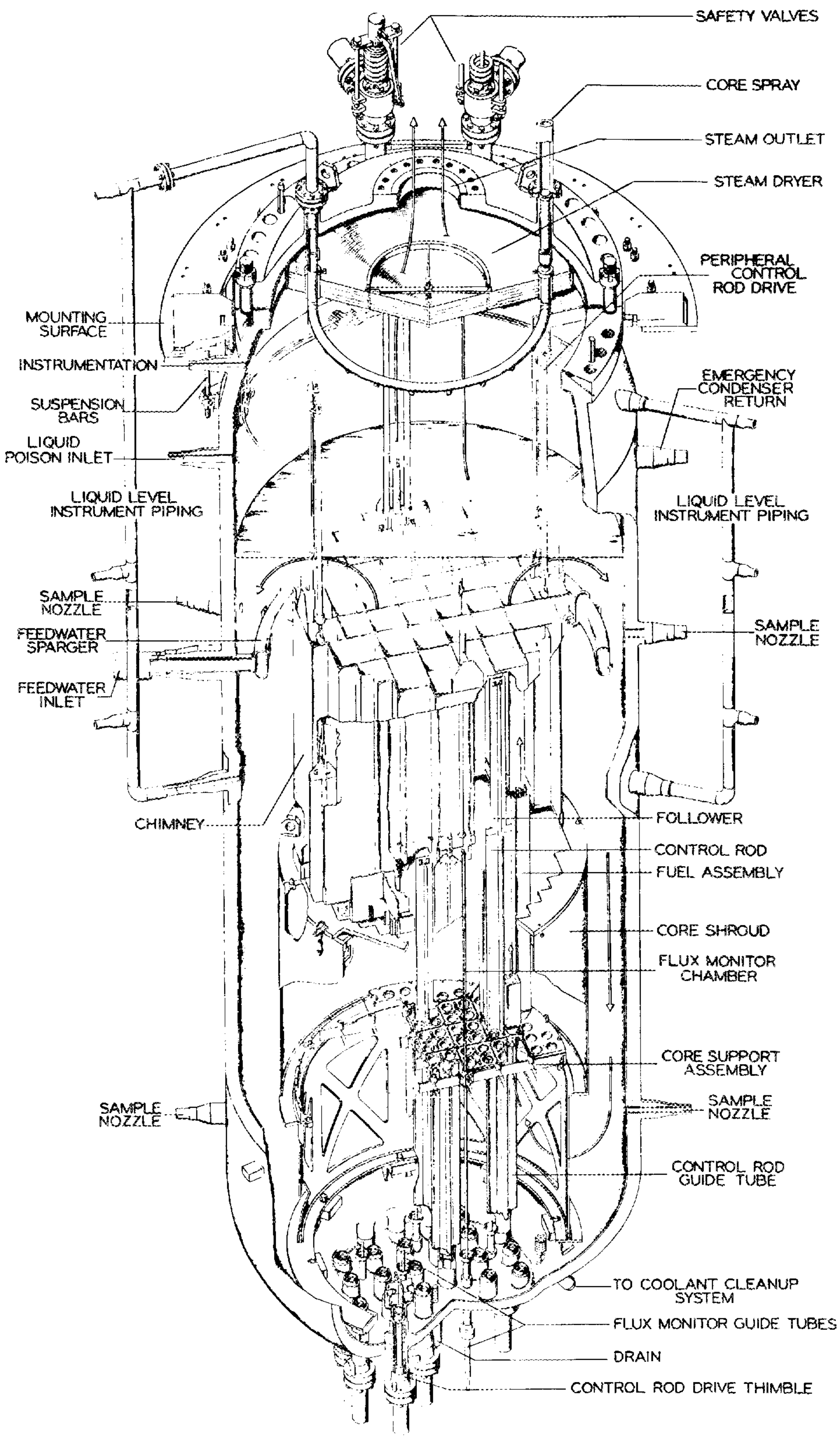



Fig. 7.56. Humboldt Bay Reactor Vessel. (From ref. 28)

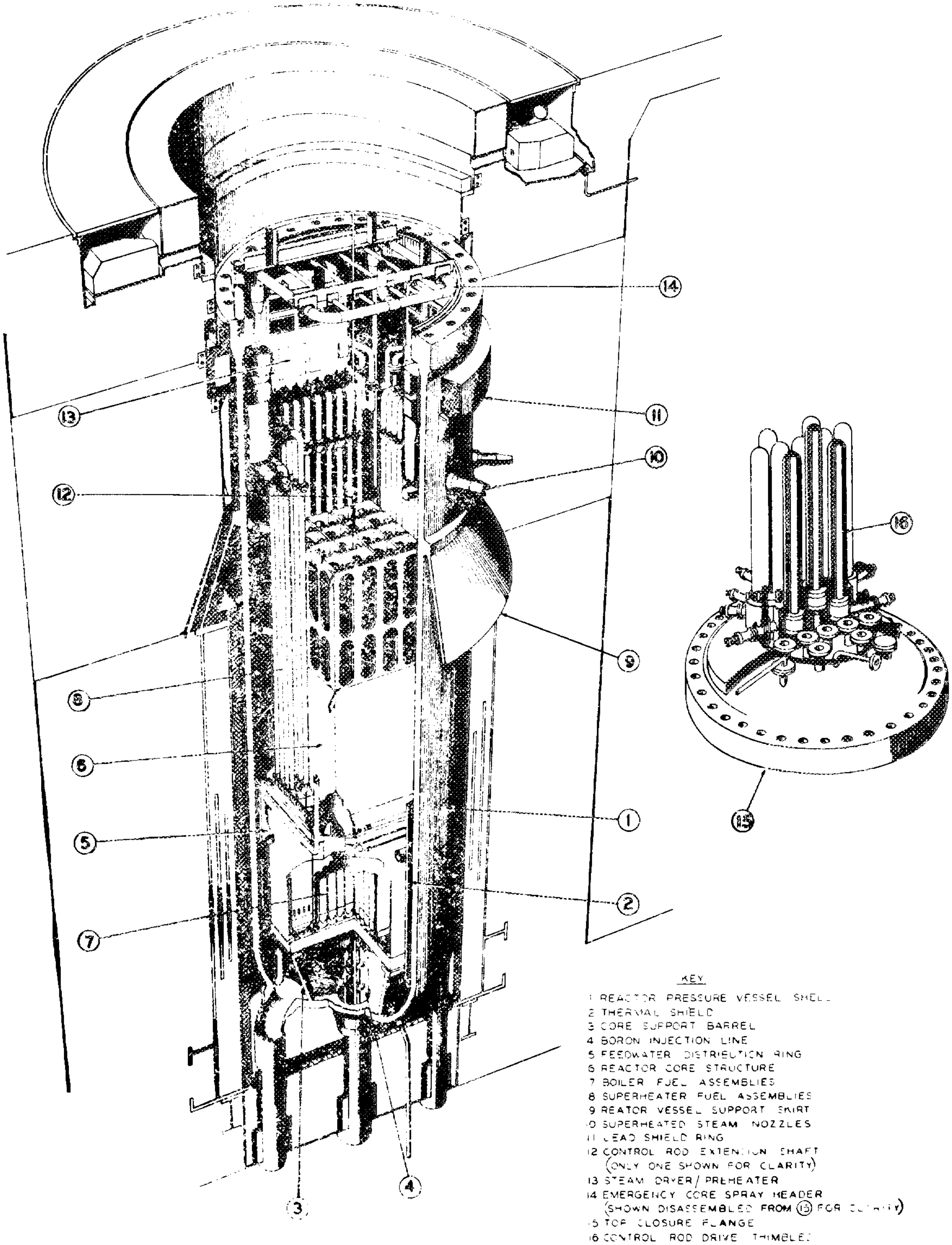



Fig. 7.57. BONUS Reactor Vessel. (From ref. 42)

forced-circulation loop would be required. It seems reasonable to design an emergency cooling loop for low-pressure operation to relieve this problem. This emergency system might use as its coolant the container atmosphere, if this were compatible with the materials in the core and moderator. If this was not satisfactory, a coolant would have to be supplied to suppress the effects of the incompatible gas. For example, the oxidation of graphite by air is a reaction which generates heat, but if enough cooling can be supplied by the air flow to offset this contribution of heat, then air is a reasonable choice for a coolant; if the converse is true, some inert diluent gas must be supplied to eliminate graphite oxidation.

The EGCR has an emergency cooling loop that employs nitrogen as a coolant.[71] (Nitrogen is employed to minimize the extent of graphite oxidation.) The loop and blowers are designed with minimum complexity and maximum reliability. The system is designed to remove all the decay heat that would be available under the worst accident conditions and to maintain an average outlet temperature of less than 1350°F. Included with this system is a filtering and release scheme, which is discussed in Section 7.9.3.

The Peach Bottom high-temperature gas-cooled reactor (HTGR) container is sealed (during operation) and filled with an inert, reduced oxygen atmosphere consisting primarily of nitrogen, with less than 5% oxygen and approximately 12% carbon dioxide by volume. Thus explosions or combustion of CO and $H_2$ (formed during a steam-graphite reaction) cannot occur. Combustion of the core graphite in the event of a primary system rupture is also effectively retarded or prevented by the reduced oxygen atmosphere. There is also a rather unique way for removing heat from the core in the case of emergency. Water is circulated through steel tubes in the emergency cooling jacket surrounding the reactor vessel. Heat is conducted and radiated from the core to the reactor vessel and then radiated and convected to the emergency cooling jacket.

### 7.9.2 Building Spray Systems

If all efforts to prevent the reactor core from melting fail, fission products may escape to the container. The released fission products will have a tendency to distribute themselves throughout the container and to follow the same paths of leakage as does the container atmosphere. The container contents will leak at a rate that is some function of the internal pressure, which is the driving force. Energy will be lost from the system by heat transfer to the ambient atmosphere, and the internal pressure will be reduced. But this may take considerable time if natural phenomena are relied upon entirely. Accordingly, internal spray and dousing systems and external spray systems would usually be employed to remove energy in the form of heat from a container after an accident. By supplying this heat sink, the internal pressure could be reduced quite rapidly. The internal spray and dousing system could also provide the additional benefit of removal of some soluble and particulate fission products. Building spray systems for several reactors are described in Table 7.63.

Table 7.63. Building Spray Systems

| Reactor | Addition or Circulation Rate | Spray Arrangement | Source and Capacity Information | Comments |
|---|---|---|---|---|
| Big Rock Point | 1000 gpm maximum addition rate<br>400 gpm recirculation rate | There are two sets of spray nozzles available; one set is automatically put into service when the sphere pressure reaches 2 psig, and the other set is put into service manually from a location outside the sphere | 100% coverage of the building with water supplied from Lake Michigan via the electric or diesel fire pumps until water level reaches a certain level in the sphere, at which time one of two recirculation pumps may be placed in service | Spray systems furnish pressure reduction after the accident; can be operated from a remote, shielded location |
| Elk River | 1000 gpm | Spray headers located above main and basement floors | 100% coverage; 30,000-gal water storage tank | Spray reduces the pressure after the accident |
| EGCR, external | 2500 gpm | Ring headers with spray nozzles; most concentrated near the top of the dome | $32 \times 10^6$ Btu/hr heat removal capacity; supplied from Melton Hill Lake | This is the primary method of energy removal and must operate after the accident |
| HWCTR | 1000 gpm for 15 min; then 120 gpm | Sprinkler heads in the dome of the container and at all levels in the containment building | Initially from storage tanks in top of container, then by pumps from wells (pressure reduction from 23 to 10 psig in 15 min) | This furnishes pressure reduction after accident; can be operated from a remote location |
| Indian Point | | | | |
| Internal | 1000 gpm | Eight spray headers in top hemisphere | Installed in such a manner as to wash down surfaces | No credit taken for internal spray cooling in mca analysis |
| External | 3000 gpm | Eight spray headers in top hemisphere | Recirculating type; the water is collected in a sump and pumped back to spray header | |
| PRTR | 500 gpm | Fog nozzles in two ring headers above main floor | Cooling power of $37.0 \times 10^6$ Btu/hr; 100% coverage of main floor | Can be operated from remote location (1/4 mile) |
| BONUS | | | | |
| Internal | 1000 gpm | Headers and nozzles over main floor and in basement | Water supplied as in Table 7.62 (pressure reduction from 4.3 to 2.4 psig in 50 min) | No credit taken in mca analysis for either system; both can be operated from remote location |
| External | 450 gpm | Two circular perforated ring headers around the upper dome | To be used to keep pressure down after initial pressure reduction by internal system; water supplied as in Table 7.62 | |
| NPD | Reactor vault: | | | |
| | 1500 Igpm | Fog nozzles suspended at two ends of reactor vault; actuated at 0.72 psig | Heavy water from moderator system | |
| | 1500 Igpm | Fog nozzles (same as above) | Supplied from standby water system | |
| | 8700 Igpm | Two perforated tanks suspended at two ends of reactor vault; actuated at 3.7 psig | Supplied from 250,000 imperial gallon storage tank[a] | |
| | Boiler room (large leaks): | | | |
| | 0 to 10 sec, 97,000 Igpm<br>10 to 20 sec, 75,000 Igpm<br>20 to 30 sec, 55,000 Igpm<br>30 to 40 sec, 36,000 Igpm<br>40 to 60 sec, 15,500 Igpm | Seven perforated tanks suspended in the boiler containment vessel; actuated at 1.5 psig | 250,000 imperial gallons storage tank[b] | |

[a]39,000 imperial gallons unavailable because of internal arrangement of the tank.

[b]Initial 100,000 imperial gallons for dousing; 150,000 imperial gallons for light-water injection.

7.9.2.1 Spray and Dousing Systems

Spray and dousing systems are provided in some reactors where there is no fear of a metal-water or graphite-water reaction occurring after the mca. They are primarily for the purpose of reducing the pressure within the containment building after the accident; consequently, the containment atmosphere after the accident will have a high water-vapor content. Therefore all electrical equipment expected to operate under postaccident conditions must be provided with proper insulation to ensure such operation; this is an added expense and may be considered a serious drawback for this type of system.

Internal sprays may also serve to scrub some fission products out of the containment atmosphere. Two important mechanisms are responsible for this action: (1) the solubility of some fission products in water (e.g., iodine, bromine) and (2) the flocculation and occlusion of particulates by the water droplet. By these actions a significant fraction of the fission products may be removed from the atmosphere and "fixed" inside the container.

As was previously mentioned, the primary purpose of introducing a water spray would be to reduce the pressure in the container. If the primary coolant system of a water reactor ruptured, a large amount of stored energy would be released to the container in the form of heat. It has been shown that the presence of water in stagnant pools reduces the containment pressure far below the calculated (adiabatic) value.[72] If a fog spray displaying many times the heat-transfer surface of an open pool were introduced, the pressure-reducing effect would be even greater.

The BONUS Reactor building is equipped with an internal spray system (also an external one), as shown in Fig. 7.35. A system of headers and nozzles located in the upper dome of the container can distribute, in a fairly uniform manner, 1000 gpm of spray water. Using this as the only heat loss mechanism, the pressure could be reduced from 4.3 to 2.3 psig in 50 min.

The NPD, a Canadian reactor, depends greatly upon this sort of "spray-suppression" system for its containment; and, as can be seen from Table 7.63, a large amount of water could be deluged into the containment structure in a very short time (92,700 imperial gallons in 3 min).

Some of the reactors have remote operating bunkers so that spray systems can be continuously operated long after the accident is over. Figure 7.58 (ref. 14) shows the location of such a bunker at the HWCTR site.

7.9.2.2 External Spray Systems

External spray systems are used to remove heat from the containment vessel and thus limit the internal pressure. In a liquid-metal-cooled or a high-temperature graphite-moderated reactor, the introduction of water into the containment vessel where it could intimately associate with the coolant or moderator could be catastrophic; therefore an external means of cooling without introducing water or its vapor to the containment atmosphere must be provided. External spray cooling is also provided when it would be prohibitively expensive to have complete spray coverage in all the compartments of a complex container. If the amount of heat that must be removed from the container to limit the internal pressure can be removed

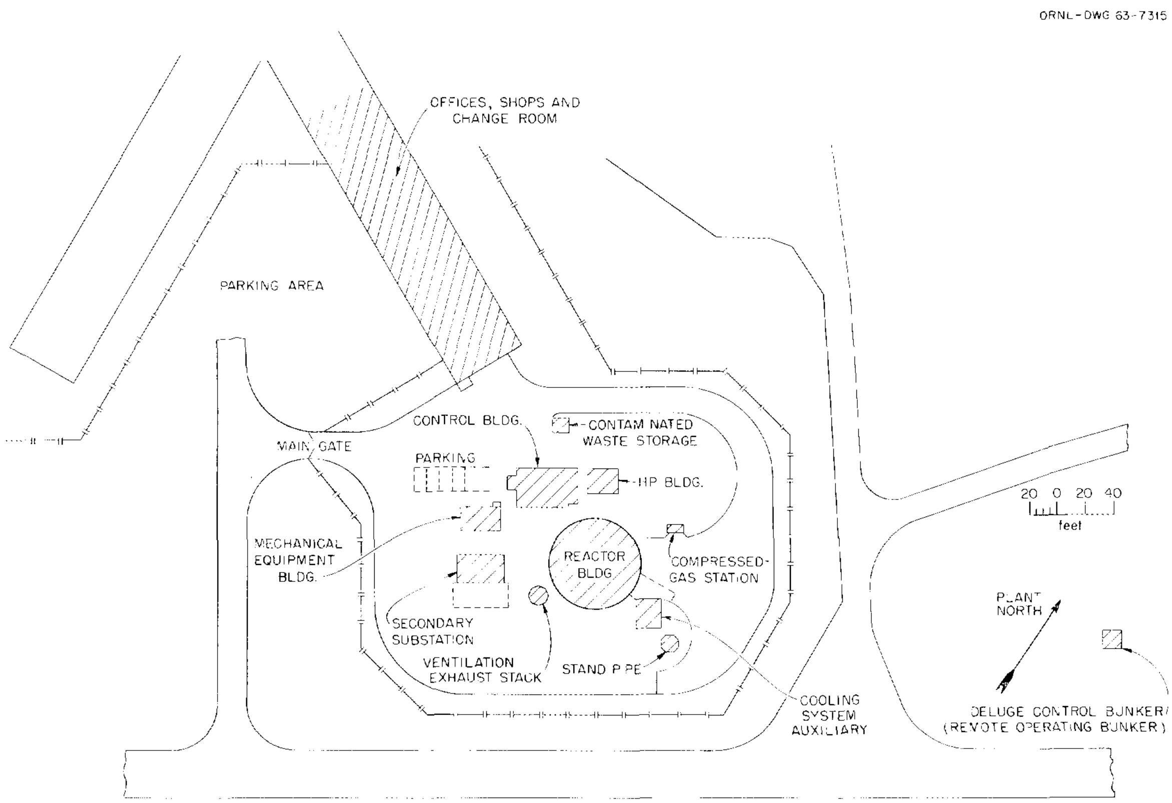



Fig. 7.58. HWCTR Plant Area Showing the Remote Operation (Emergency) Station. (From ref. 14)

by either an internal water fog or by heat transfer to a cold surface, the designer may choose between the two. However, if the cooling of a surface by an external spray is cheaper and satisfactory, the external spray should be used. Furthermore, if an internal spray were used, the electrical equipment within the containment vessel would have to be provided with the proper insulation to operate in an atmosphere with a high water-vapor content, and this would be reflected in increased plant costs. The selection of the EGCR external spray was based on the preceding arguments.

The BONUS and Indian Point, as well as EGCR, plants have external spray schemes in effect. The BONUS and Indian Point external sprays would be used in conjunction with internal spray systems; but the EGCR would depend on the external system for all its induced cooling of the containment building. At the EGCR, water would be introduced by ring headers near the top of the containment dome and then allowed to run down the sides of the container to cool its surface.

### 7.9.3 Filter Systems

Filters for the removal of iodine and particulates are provided for cleaning the containment atmosphere before a substantial amount of radioactivity leaks out or before treated air is released to the environment. Since the most severe effect of an accident is the dispersion of dangerous fission products, this scheme of fission-product removal may become the most important "engineered safeguard." Much development work has been done in this area, and even more is needed. The potential importance of filter systems has been demonstrated by several nuclear power companies by the fact that these systems have been installed in the newer plants, even though no credit has been taken in reducing the effect of the maximum accident. Table 7.64 describes the filter systems installed in several typical containment structures.

Some plants take credit for the mitigating effect of filter systems. In order for these plants to claim any benefit from the filters, the reliability of the overall system must be demonstrated, as well as the efficiency of the filtering system under the expected accident atmosphere. Provision must be made for the unexpected, such as fires in charcoal filters initiated by decay heat of fission products, attrition of the charcoal beds, commonly called "dusting," and blanketing of charcoal filters or metallic absorbers with water vapor, which may reduce the efficiency. The efficiency of any iodine removal system depends strongly upon the chemical form of the iodine to be removed.

Two important considerations are the general reliability of the blowers, filters, filter housings, seals, etc., and the relative invulnerability of the system to damage from particles, missiles, chemical reagents, vapors, shock waves, etc. Furthermore, adequate cooling must be provided for filters, as well as other absorbers, to remove decay heat from fission products that are collected. It may also be necessary to shield the filters (or absorbers) if a significant amount of fission-product activity is expected to be collected.

There are, in general, two types of filtering systems in use at the present time: the recirculating system and the once-through system. The only difference in these two systems is what is done with the filtered air. In the recirculating system, it is discharged back to the containment vessel, thus reducing the concentration of the activity in the containment vessel by an amount that is a function of the filter system performance. The once-through system discharges the filtered air directly to the environment, usually through a stack. This discharged air is closely monitored for fission-product concentration.

The system is usually installed within the containment shell, where blowers induce air movement through the filter system. As may be seen in Table 7.64, the filter units usually consist of a prefilter followed by an absolute particulate filter that is followed by an iodine-removal section consisting of activated charcoal or silver-plated wire mesh. The HWCTR, for example, uses four separate systems, each with its own independent blowers, motors, and filters. When all are operating, they can filter 4000 cfm, which provides for the filtering of one containment volume every 1.3 hr.

A different approach to the use of filters is demonstrated by the Indian Point and NS Savannah systems. In these cases, the containment shell is housed in a compartment or building, and thus any leakage from the containment vessel is held up in the second compartment. It is the atmosphere of this second compartment that is filtered and fed to the stack. This concept is shown schematically in Fig. 7.59.

Table 7.64. Filter Systems

| Facility | Circulation or Discharge Rate (cfm) | Filter Function | Filter Description | Required Performance at mca |
|---|---|---|---|---|
| CVTR | 1000 | Recirculation for particulate and iodine removal | Prefilter, absolute filter (99.97% removal of 0.3-μ particles), and iodine-removal section of activated charcoal (1 in. deep, 90% iodine removal); two such units, one in standby | No credit taken in mca analysis |
| EGCR (1) | 750 (max) | Once through; discharge to atmosphere | Prefilter (95% removal of 5-μ particles), absolute filter (99% removal of 0.3-μ particles), charcoal trap (95% iodine removal), and absolute filters (99% removal of 0.3-μ particles) | 95% removal of halogens and 99% of particulates |
| EGCR (2) | 41,660 | Recirculation system | Absolute (99.97% removal of 0.3-μ particles); silver-plated copper mesh for iodine removal | No credit taken in mca analysis |
| HWCTR | Four units of 1000 each | Recirculation for particulate and iodine removal | A unit contains a moisture separator and a bed of activated (coconut shell) charcoal (56 lb) shown to remove iodine with an efficiency of >99.9% | Credit is taken for one of the four units operating at stated efficiency (primarily for iodine removal) |
| Indian Point | 10,000 | Once through for particulate removal | Prefilters; airborne activity filters (99.97% removal of 0.3-μ particles) | No credit taken in mca analysis |
| NS Savannah | Two units (1000 each), one in standby | Once through discharge to atmosphere; system for particulate and iodine removal | Prefilter, absolute filter (99.95% removal of 0.3-μ particles), and iodine-removal section consisting of silver-plated copper mesh (6 in. thick), activated charcoal (1 in. effective thickness), silver-plated copper mesh (6 in. thick) | 99% removal of iodine with infinite cloud release is assumed |
| Yankee | 12,000, three units each of 4000 capacity | Recirculates container air to remove particulates | Prefilters, airborne activity filters (99.95% removal of 0.3-μ particles) | No credit taken in mca analysis |
| NPD (1) | 32,000 | Recirculation through reactor vault | $D_2O$ sprays, filters, and ion exchange[a] | |
| NPD (2) | 1000 | Recirculation from boiler room to remove particulates | Filter (99.97% removal of 0.3-μ particles)[a] | |
| Humboldt Bay (refueling building) | 134 | Once through; for particulates and iodine removal | Caustic scrubber and "absolute filters" (95% minimum removal of halogens and particulates) | |
| BONUS | 50,000 (building ventilation) | Once through for particulate removal | Oil-treated prefilter and glass fiber filter | |
| Elk River | 3500 (max) | Once through for particulate removal | Prefilter, absolute filter (99.95% removal of 0.3-μ particles) | |
| Peach Bottom | Two units; 2000 cfm each; one in standby | Recirculation for strontium removal | Absolute (99.97% removal of 0.3-μ particles) | 50% removal of strontium |

[a] In emergency, vented to stack through activated charcoal and absolute filters.

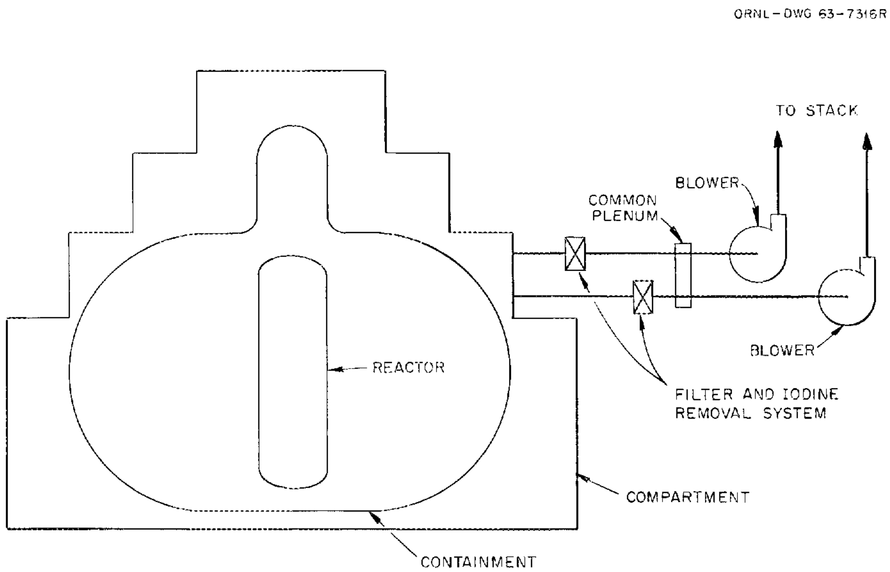



Fig. 7.59. The NS Savannah Containment Concept.

### 7.9.4 Multiple Containment

The concept of the double containment barrier was brought into focus recently by the proposed Ravenswood Plant of the Consolidated Edison Co. (see Sec. 7.8). This constitutes an engineered safeguard and would, alone, serve as added protection for the public from an accident within the container. The basic idea is that of a container which is contained. The second volume, the annular space formed around the first container (see Fig. 7.53), serves as a holdup and dilution volume for fission products that leak from the inner container (see Sec. 4.4.4). Aside from the cost, there is no reason that a third container could not be constructed that would thus provide more holdup and more dilution of the fission products. This could, of course, be extended to many containers, but it is more reasonable to use the "two-container" concepts in conjunction with other engineered safeguards (see Sec. 7.9.5).

As an example of this concept in action, the Indian Point plant of Consolidated Edison is housed in a conventional steel sphere that is in turn housed in a concrete building (see Fig. 7.9). If the concrete building were not there, the doses from $I^{131}$ activity would be greater than those set forth in the AEC Site Criteria, but the building provides sufficient holdup and dilution to bring the doses well within the AEC guide lines.

### 7.9.5 Combination of Engineered Safeguards

As was mentioned in the previous section, multiple containment alone is a concept that yields significant advantages, but when used in connection with other safeguards, the effectiveness and reliability may be increased, with a reduction in capitol cost. Most, if not all, reactors (with the possible exception of liquid-metal or molten-salt reactors) have some provision for emergency core cooling; therefore any other engineered safeguard makes that a combination system. For example, the EGCR has emergency core cooling, container cooling for pressure reduction, and filtering to remove fission products. Proposals have been made for containers with spray cooling for pressure reduction and the multiple barrier for fission-product control. For the Ravenswood plant, multiple containment with filtering of the air drawn from the annular space (Fig. 7.53) is proposed.

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# 8. STRUCTURAL DESIGN CONSIDERATIONS

A. M. Appleford* J. R. Hawthorne**
G. H. Dyer* H. B. Piper***
R. F. Griffin* L. E. Steele**

The preceding chapters of this report discuss the need for reactor containment and present the theoretical considerations and analytical techniques from which the basic containment requirements of design pressure and maximum allowable leakage rate are established. Various types of containment are described and information is presented on many specific containment systems. This chapter serves to bridge the gap between the basic containment requirements and the completed containment system by presenting some major practical considerations involved in translating the requirements into an efficient, economic and reliable containment system suitable for a specific situation.

Some of the factors considered in determining the optimum containment type, size, shape, and material to meet the basic containment requirements are discussed, and attempts are made to indicate why some containment designs are suitable for some situations and not for others. Alternate techniques of containment construction are also discussed, and the most important steps of each are outlined.

Throughout this chapter, the details and the problem areas unique or most significant to containment structures are emphasized. There is only limited discussion of those areas which, although they may be of major importance, are common to the engineering and construction of conventional structures.

## 8.1 GENERAL DESIGN CONSIDERATIONS

Design pressure and maximum allowable leakage rate are the basic design requirements specified for most containment structures. However, particular containment designs may require that other special design requirements be specified, such as the venting rates for pressure-relief and pressure-suppression containment. The bases on which these requirements are established are discussed in Chapters 4 and 6. In determining how to best meet these requirements, the containment designer must consider many factors in arriving at an optimum containment design for a specific situation. Many of these factors and the effect they may have on the selection of containment type and on containment design are discussed in this section. Underlying all these considerations is the objective of providing an effective containment structure as economically as possible. The economics of containment are discussed more specifically in Chapter 11.

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*Bechtel Corporation, San Francisco, Calif. These authors prepared all but Section 8.5 of this chapter.

**U.S. Naval Research Laboratory.

***Oak Ridge National Laboratory.

### 8.1.1 Site Conditions

The location of a reactor plant with respect to populated areas and load centers will have a considerable influence on the degree of containment required, as discussed in Chapter 1, and on overall plant economics. In addition, local site conditions will influence the type of containment selected and may dictate against the use of an otherwise favorably located site.

#### 8.1.1.1 Topography and Soil Conditions

The soil conditions at the plant site are important factors in determining the type of foundation to be used and in deciding whether to use an aboveground or an underground design. For example, rock and densely compacted soil favor aboveground containment designs because of high excavation costs. Dense, uncompacted soil with no rock and a ground-water table lower than the lowest excavation are suitable for some forms of underground construction. Where a higher ground-water table exists, together with granular soil containing no hard materials or underground obstructions, caisson construction techniques may be suitable, as discussed below. If underground obstructions, hard materials, and a suitable subsurface stratum are present with a high ground-water table, a containment design compatible with the use of pilings in the foundations should be selected.

Rock and dense soil have good load-carrying capability. The unreinforced saucer-shaped foundation normally used under spherical and cylindrical containment structures is readily adapted to such site conditions. Other aboveground containment structures that do not require much excavation are also suited to these ground conditions. Sites with dense soil containing little or no rock and a low ground-water table are the most suitable for underground containment structures requiring dry conditions for erection, since excavation costs are minimized with these conditions.

Granular soil and a high ground-water table may favor an underground containment structure built using caisson construction techniques, as illustrated in Figs. 8.1 through 8.5. A caisson is sunk into the ground in such soil by using a clamshell or similar excavating equipment to remove ground material through the caisson. The watery, granular soil conditions, assisted by water jets, create a flow of material into the excavated region from around the caisson periphery. As this material flows toward the excavation from under the caisson walls, the caisson sinks into the ground. If hard materials or underground obstructions are present, they will increase the cost and may even prevent the sinking of caissons, so other designs should be considered.

Soft soil with a high ground-water table and subsurface hard strata or obstructions are conditions for which pile foundations should be considered. Flat-bottomed containment structures are most easily built on pile foundations (Fig. 8.6). A sphere or a cylinder with an ellipsoidal bottom head can be adapted to pile foundations either by varying the cutoff elevations of the pile to conform to the containment contour (Fig. 8.7) or by building a substructure on top of the pilings (Fig. 8.8).

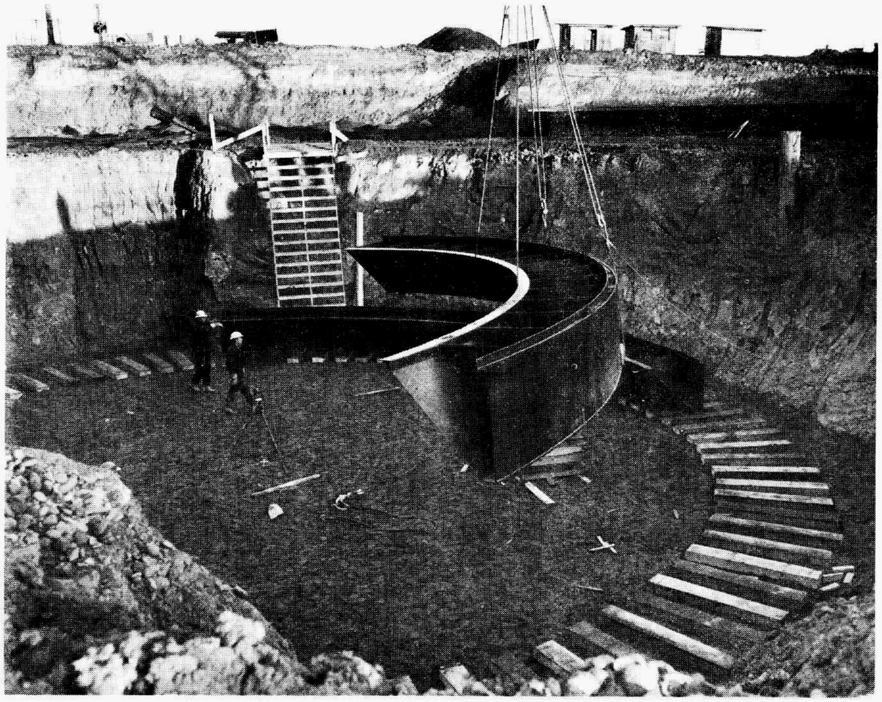

Fig. 8.1. Caisson Bottom Cutting Edge at Humboldt Bay Site.

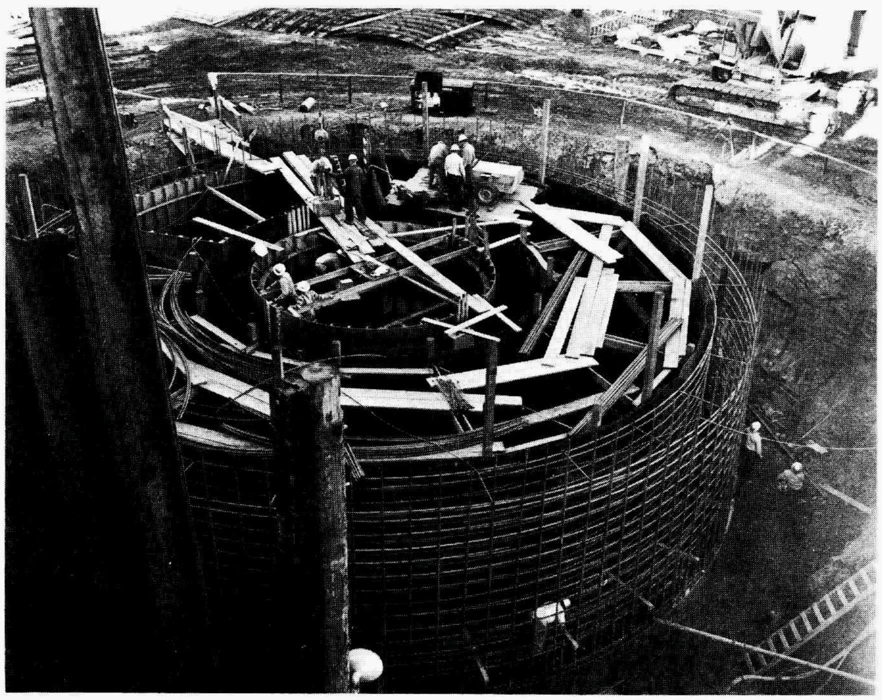

Fig. 8.2. Liners and Reinforcing Steel for Lower Lift at Humboldt Bay Site.

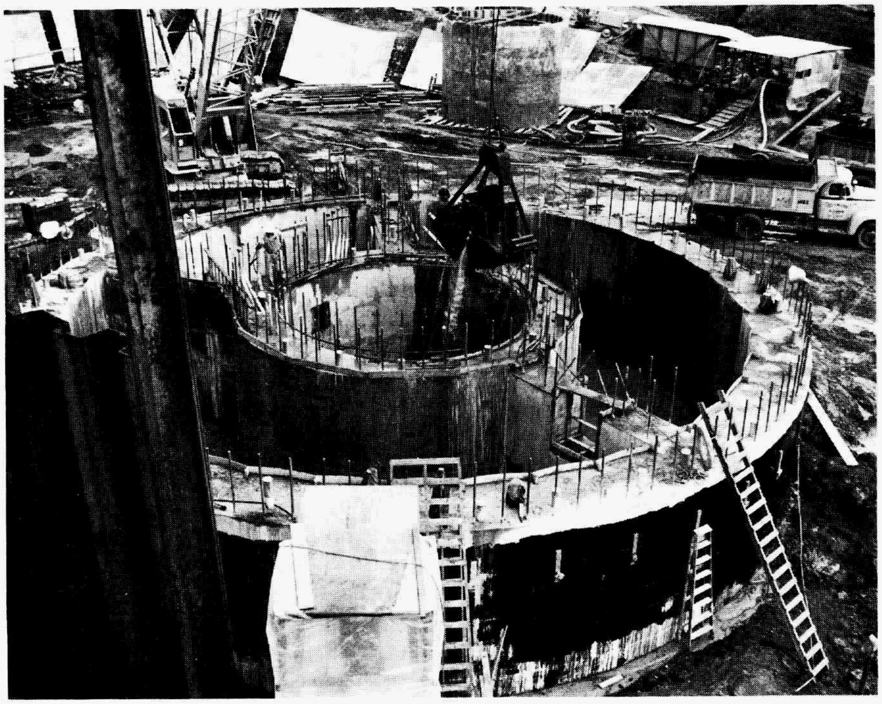

Fig. 8.3. Initial Sinking of First Lift at Humboldt Bay Site.

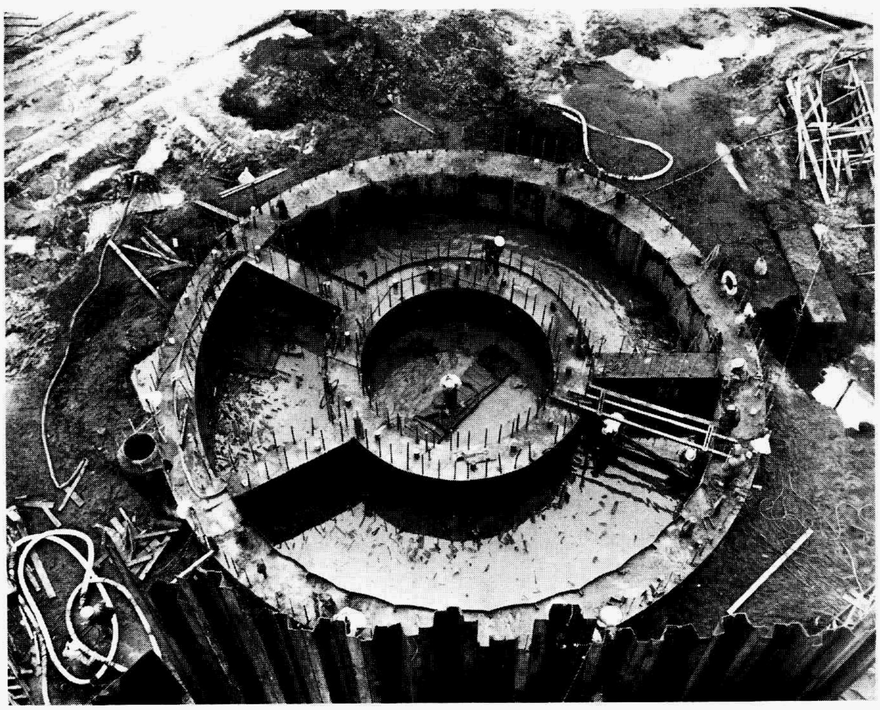

Fig. 8.4. Final Sinking of First of Six Lifts at Humboldt Bay Site.

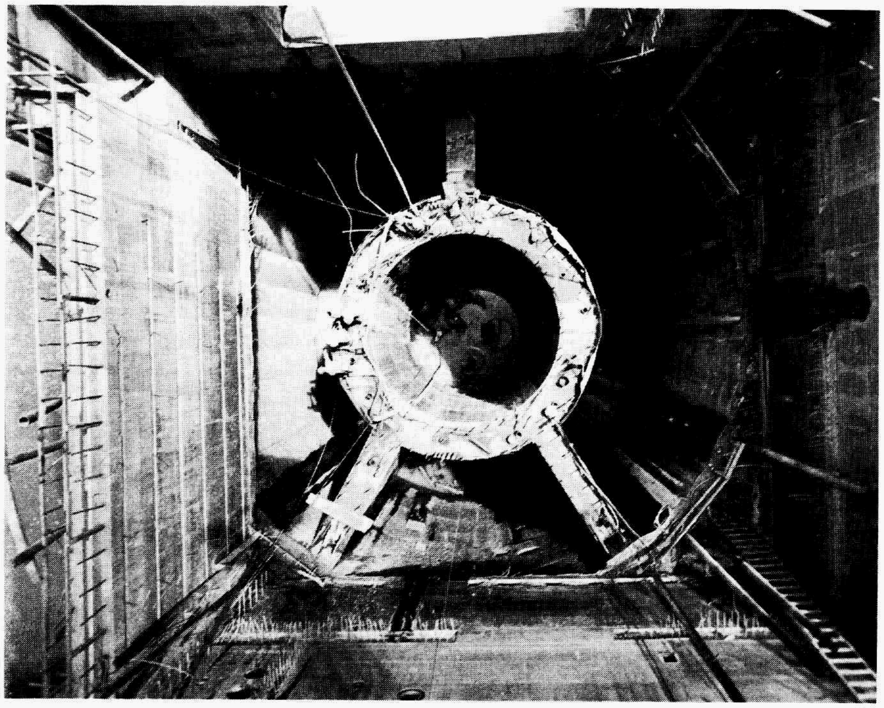

Fig. 8.5. Bottom Tremie Concrete Placed and Vessel Dewatered at Humboldt Bay Site.

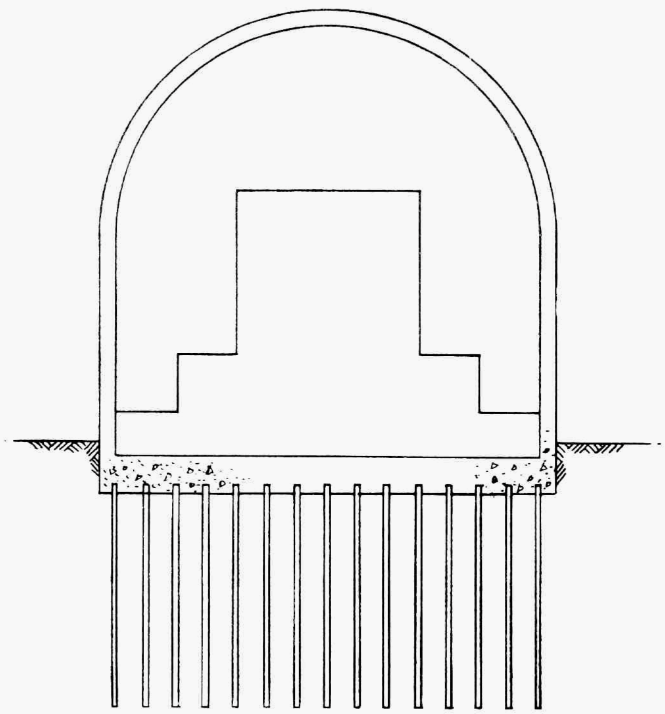

Fig. 8.6. Flat-Bottom Containment Structure on a Pile Foundation.

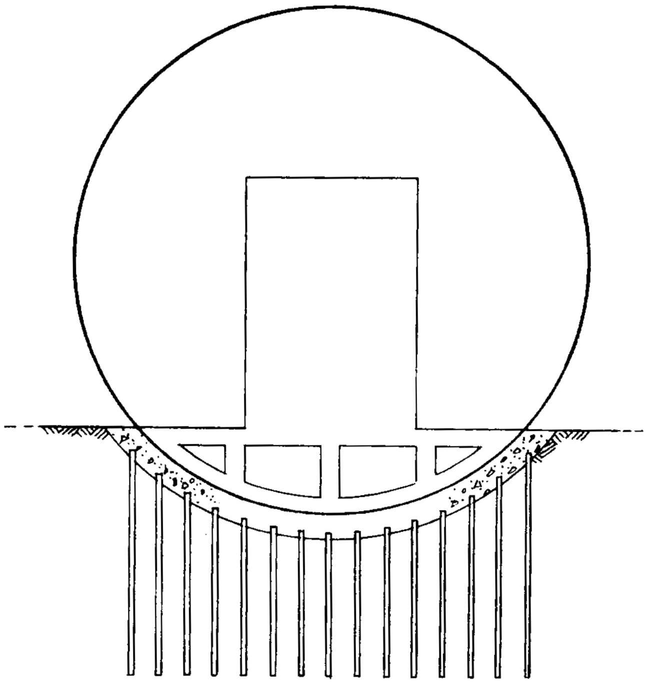

Fig. 8.7. Curved-Bottom Containment Structure with Varying Pile Cutoff Elevations.

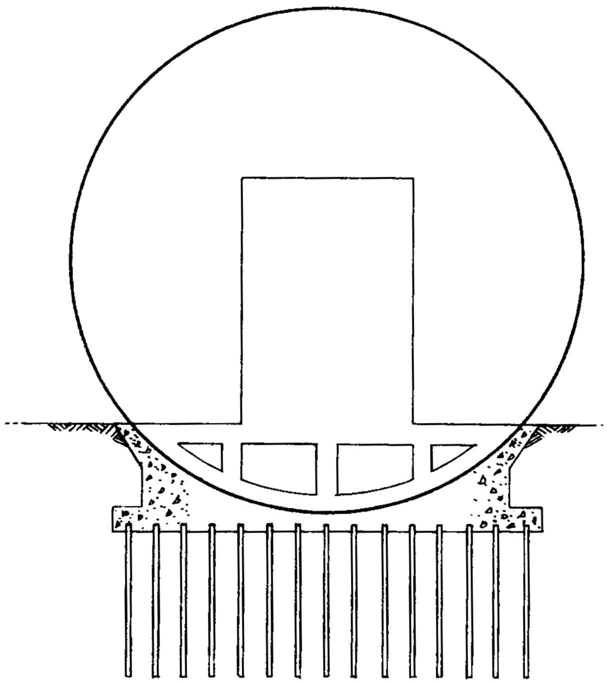

Fig. 8.8. Curved-Bottom Containment Structure with Substructure on Top of Evenly Cut Piles.

If the site has soft or granular soil and a suitable subsurface stratum, a partially underground containment structure founded on the subsurface stratum may be the most economical choice (Fig. 8.9). Either the bottom of the containment structure or the foundation may have a ring extending horizontally outside the containment walls. This ring is covered with backfill, the weight of which bears on the ring and counteracts buoyant forces on the containment structure. A high ground-water table or artesian aquifer creates a large buoyant force while decreasing the effective weight of the backfill. Under such ground-water conditions a large ring is required to prevent the containment vessel from buoyantly rising in the ground.

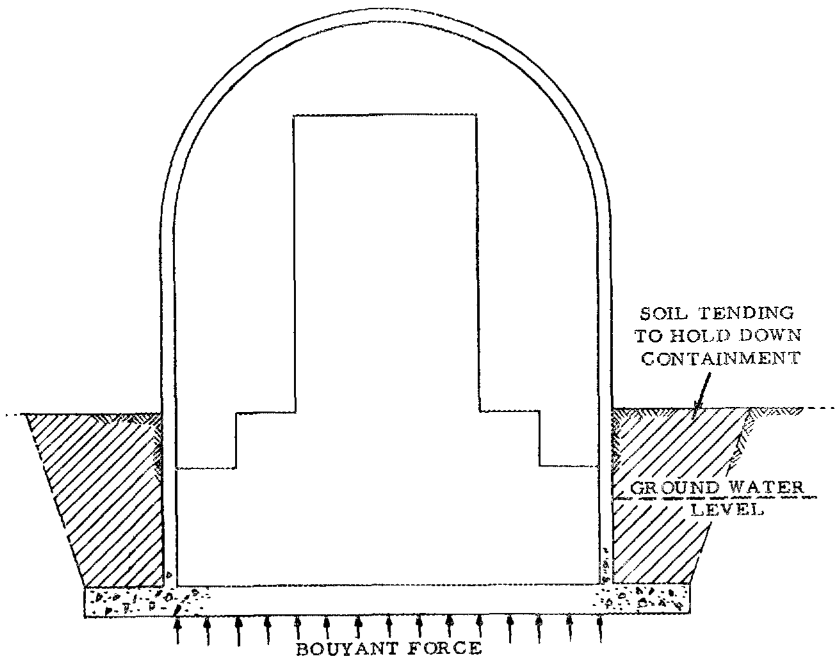


Fig. 8.9. Partially Underground Containment Structure.

8.1.1.2 Meteorology and Hydrology

The design studies of aboveground containment structures must include consideration of the meteorological phenomena that will affect both the dispersion of uncontained radioactive materials and the structural design, the latter in a manner similar to more conventional structural design considerations. The containment system must meet the design leakage rate requirement, which is partially dependent on site meteorological conditions, and it must be designed for and protected from weather phenomena such as rain, snow, wind, and thermal conditions that might be expected to occur during the life of the system. In some cases, the structural design requirements imposed by these naturally occurring conditions may be more significant than those imposed by the pressure and temperature conditions postulated under the maximum credible accident conditions.

Although the specific local codes legally in force should be consulted, the probable local requirements may be estimated by the use of the following information extracted from the Uniform Building Code,[1] which has been adopted as the legal code of many local jurisdictions:

Sec. 2303 (in part)

"Wind and earthquake loads need not be assumed to act simultaneously."

Sec. 2305 (in part)

"Where snow loads occur, the roof structure shall be designed for such loads as determined by the Building Official."

Sec. 2307 (in part)

"(a) General. Buildings or other structures shall be designed to withstand the minimum horizontal pressures set forth in Table No. 23-C [Table 8.1], allowing for wind from any direction. The wind pressures set forth in Table No. 23-C are minimum values and shall be adjusted by the Building Official for areas subjected to higher wind pressures. When the form factor, as determined by wind tunnel tests or other recognized methods, indicates vertical or horizontal loads of lesser or greater severity than those produced by the loads herein specified, the roof structure may be designed accordingly.

Table 8.1. Recommended Minimum Allowable Wind Pressures for Various Height Zones Above Ground[a]

| Allowable Resultant Wind Pressure at 30 ft Above Ground[b] (psf) | Minimum Allowable Resultant Wind Pressure (psf) at Indicated Height[c] | | | | | |
|---|---|---|---|---|---|---|
| | <30 ft | 30 to 49 ft | 50 to 99 ft | 100 to 499 ft | 500 to 1199 ft | 1200 ft and over |
| 20 | 15 | 20 | 25 | 30 | 35 | 40 |
| 25 | 20 | 25 | 30 | 40 | 45 | 50 |
| 30 | 25 | 30 | 40 | 45 | 55 | 60 |
| 35 | 25 | 35 | 45 | 55 | 60 | 70 |
| 40 | 30 | 40 | 50 | 60 | 70 | 80 |
| 45 | 35 | 45 | 55 | 70 | 80 | 90 |
| 50 | 40 | 50 | 60 | 75 | 90 | 100 |

[a]Based on Table 23-C of ref. 1, p. 102.

[b]See Fig. 8.10. Row in table should be selected that corresponds to the allowable resultant wind pressure indicated for the particular locality being considered.

[c]These requirements do not provide for tornadoes.

"(b) Wind Pressures. Roofs of all buildings or other structures shall be designed and constructed to withstand pressures, acting upward normal to the surface, equal to one and one fourth times those specified for the corresponding height zone in which the roof is located. The height is to be taken as the mean height of the roof structure above the average level of the ground adjacent to the building or other structure and the pressure assumed on the entire roof area.
"(c) Roofs with Slopes Greater than 30 Degrees. Roofs or sections of roofs with slopes greater than 30 degrees shall be designed and constructed to withstand pressures, acting inward normal to the surface, equal to those specified for the height zone in which the roof is located, and applied to the windward slope only.

..............................................................

"(e) Anchorage Requirements. Adequate anchorage of the roof to walls and columns, and of walls and columns to the foundations to resist overturning, uplift, and sliding, shall be provided in all cases.
"(f) Solid Towers. Chimneys, tanks, and solid towers shall be designed and constructed to withstand the pressures as specified by this section, multiplied by the factors set forth in Table No. 23-D [Table 8.2].

..............................................................

"(i) Moment of Stability (Design). The overturning moment calculated from the wind pressure shall in no case exceed two-thirds of the dead load resisting moment.

The weight of earth superimposed over footings may be used to calculate the dead load resisting moment.
"(j) Combined Wind and Live Loads. For the purpose of determining stresses all vertical design loads except the roof live load and crane loads shall be considered as acting simultaneously with the wind pressure."

In tornado areas, or in other areas of abnormal or especially severe wind conditions, additional special provisions may be required. Figure 1 and

Table 8.2. Multiplying Factors for Obtaining Allowable Wind Pressures on Chimneys, Tanks, and Solid Towers[a]

| Horizontal Cross Section | Multiplying Factor |
|---|---|
| Square or rectangular | 1.00 |
| Hexagonal or octagonal | 0.80 |
| Round or elliptical | 0.60 |

[a]From ref. 1, p. 103, Table 23-D.

Tables 23-C and 23-D from The Uniform Building Code are reproduced here as Fig. 8.10 and Tables 8.1 and 8.2.

Underground containment systems must also have a maximum allowable leakage rate commensurate with the permeability of the soil, the ground-water velocity and direction, and the proximity of the containment system to facilities ultimately involved with human activity. Underground systems must be designed to withstand the ground-water, soil pressure, and subsurface corrosion conditions. Again, the normal structural requirements of underground containment systems may exceed those imposed by the pressure and temperature conditions postulated under the maximum credible accident.

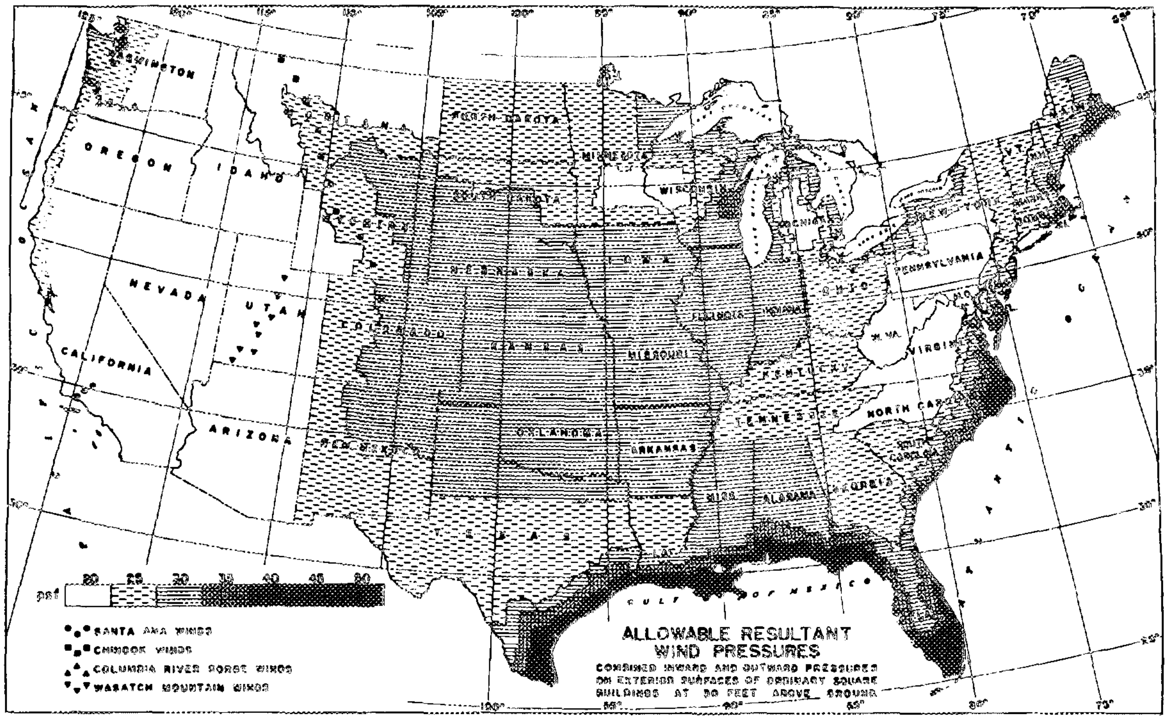


Fig. 8.10. Allowable Resultant Wind Pressures in the United States. (From Uniform Building Code, ref. 1, Section 2307, p. 102, Fig. 1)

### 8.1.1.3 Seismology

The degree of seismic activity is a site characteristic of particular concern in the design of containment structures. During an earthquake, intense lateral loads can be imposed on a containment structure at the same time that there is also an increased possibility of damage to the reactor system, so the integrity of the containment system is particularly important. In areas of high seismic activity, it is essential that the possible effects of earthquakes on the containment structure be assessed and that suitable design factors be used that will assure a very low probability of damage from earthquakes. Even in areas of relative seismic stability, consideration should be given to possible lateral earthquake loads on containment structures. In some instances, selection of the site may be dependent on the earthquake hazard. Section 100.10.C.1 of Part 100, Reactor Site Criteria, Title 10, Code of Federal Regulations, states that a reactor should not be located within 1/4 mile of a known active fault line.

The nature of earthquakes, their causes, and their probable occurrences are matters that are beyond the scope of this report. It is therefore important that a seismologist who is thoroughly familiar with the characteristics of the site being considered be employed to provide a seismic analysis, including determination of the expected probable maximum lateral earthquake load intensity.

During an earthquake, the base of a structure is moved by the ground both horizontally and vertically. The horizontal movement is usually the more intense component, and it generally causes more severe structural responses than does the vertical ground motion. The vertical component usually can excite only inconsequential oscillations in very tall structures. Only in special circumstances is consideration of this vertical motion required for containment systems.

The base of a structure built on firm ground will move with the ground during an earthquake. The principal design consideration is to assure that the structure can safely withstand this lateral acceleration, or apparent force, which is commonly expressed as a fraction of the gravitational acceleration, g. The overall response of a structure to an earthquake motion of a given intensity depends upon the impedence matching of layers of earth, the structure's vibrational characteristics, and its inherent damping qualities. Rigid structures essentially move with the exciting ground motion, experiencing the same lateral acceleration. Containment structures are typically rigid, having undamped natural vibrational periods less than 0.5 sec, and should be designed to safely withstand the probable maximum seismic ground acceleration. The aseismic design of containment structures should include (1) determination of the maximum probable lateral earthquake force, (2) effects of actual earth movements, (3) calculations of the undamped natural frequency of the structure, its internals, and adjacent structures, (4) spatial arrangement or bracing of the containment structure and other structures to prevent undesirable interaction during earthquakes, (5) determination of the required strength of the containment vessel to safely withstand base shear and moment forces resulting from seismic loads, (6) investigation of the restoring moment of the containment structure and its foundation to resist overturning effects, and (7) design of the containment shell to resist possible horizontal torsional moments.

Paragraph N-1111 of Section III of the ASME Boiler and Pressure Vessel Code[2] incorporates paragraph UG 22 of Section VIII (ref. 3) which defines the loadings, including seismic loadings, that must be considered in the design of a containment vessel. However, rules for design are not covered specifically within the ASME Code. The aseismic design must be checked against the provisions of applicable local building codes. Local codes establish legal minimums that must be provided for in designs. The Uniform Building Code[1] is a widely accepted, typical example of such a document. In general, it requires that structures be designed to safely withstand seismically induced base shear and moment forces that depend primarily on the total mass and undamped natural frequency of the structure. Seismic forces may be modified, according to the Uniform Building Code, in recognition of a structure's intended usage and the seismicity of the location under consideration.

The Humboldt Bay plant was built in an area of relatively high seismic activity. Over the last 275 years, nine shocks occurred that resulted in maximum ground accelerations of 0.25 g at the plant site. Consequently, the containment system was constructed to safely withstand a 0.25-g lateral acceleration. The Uniform Building Code would require ability to withstand a maximum lateral design acceleration of 0.15 g in this instance.

For conceptual studies of containment structures, the earthquake hazard for a particular location can be estimated from information presented in the Uniform Building Code and from an estimate of the probable maximum lateral acceleration. Figure 8.11, taken from the Uniform Building Code, indicates zones in the United States of approximately equal seismic probability, and the following tabulation presents coefficients, Z, for modifying the probable maximum lateral acceleration in each zone in consideration of the frequency of earthquakes of engineering significance:

| Zone | Z, Zone Factor |
|---|---|
| 0 | 0.25 |
| 1 | 0.25 |
| 2 | 0.5 |
| 3 | 1.0 |

In the absence of more exact information, the probable maximum lateral acceleration may be assumed to be 0.33 g. This represents the strongest ground acceleration yet recorded in the United States (El Centro, California, May 18, 1940) and is usually considered to be the probable maximum lateral earthquake force in highly seismic areas. Thus the maximum base shear force in a containment vessel due to earthquake loading is the product of the zone factor given above, the acceleration of 0.33 g, and the mass of the structure. Maximum bending moments can be computed in accordance with the Uniform Building Code. While this procedure may not provide an accurate indication of the maximum earthquake forces on a structure in a particular area, in most cases it will provide a conservative estimate and is considered adequate for conceptual design.

An AEC report, "Nuclear Reactors and Earthquakes,"[4] presents a more thorough discussion of the effects of earthquakes on nuclear power plants. This reference should be consulted for additional information on seismology and for more exact earthquake design procedures. Housner's paper, "Design of Nuclear Power Reactors Against Earthquakes,"[5] presents similar information. There is, at present, no generally applicable code for specifying the earthquake resistance of nuclear reactor containment systems, but work is proceeding in the American Nuclear Society, under task number ANS-7.72, Earthquake Design Criteria, of subcommittee ANS-7, Nuclear Reactor Components, to define this area more accurately and possibly, in the future, to issue a standard covering seismic design of containment structures.

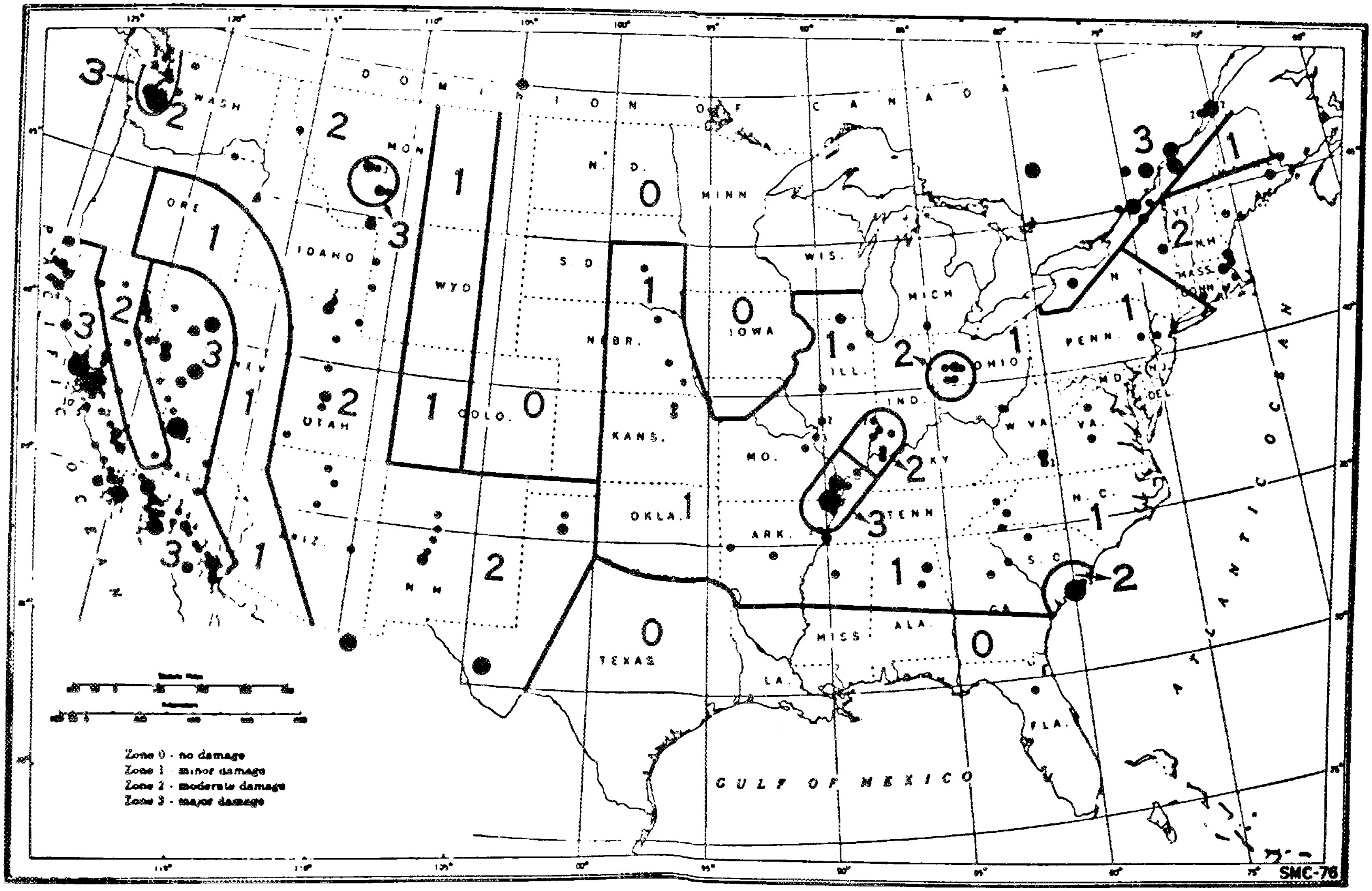


Fig. 8.11. Map Showing Zones of Approximately Equal Seismic Probability. (From Uniform Building Code, ref. 1)

### 8.1.2 Containment Envelope Boundaries

For solid fuel reactors, the relative quantities and concentrations of radioactive materials that must be contained are largest within the envelopes formed by the fuel cladding and decrease with the successive surrounding coolant fluid and containment envelope boundaries. The sources of these radioactive materials consist both of materials generated or originally placed within that particular envelope boundary and those that may have escaped from a preceding envelope containing a higher concentration of radioactivity. Failure of any boundary to contain these materials will result in their release to a subsequent envelope or to the environment. Therefore, a containment system must encompass all areas in which a significant quantity of radioactive material is placed or generated, including all preceding higher concentration envelopes that may be subject to failure. By definition, "significant" means that amount which, if released to the atmosphere, could cause radioactive material concentrations higher than those allowed by existing regulations.

The probability of failure of any particular component or barrier is normally much larger for stressed or working components than it is for unstressed or static components. Since the containment system is the last barrier to the uncontrolled release of significant quantities of radioactive materials, its boundaries are preferably placed outside the influence of process-induced stresses where, except for accident loadings, it is subject only to normal structural and environmental stresses.

The considerations discussed above tend to define the boundaries of the containment system and the specific items of equipment which must be enclosed. This equipment always includes the reactor core, reactor vessel, and primary shielding. It may or may not include the entire primary cooling fluid circuit, additional shielding, numerous auxiliary facilities, electric power generation equipment, and spent fuel handling and storage facilities.

The extent to which the primary system is contained will depend on the type of reactor, the economics of including various components, the degree of radioactive material contaminant allowed in the fluid during normal operation, and the degree of radioactive material release allowable for the site under consideration. Thus, the containment system for an indirect-cycle pressurized-water reactor usually encloses the entire primary coolant loop and related equipment, but the containment system for a direct-cycle boiling-water reactor does not usually enclose the main turbine-generator equipment, which is an integral part of the primary coolant loop. However, such a system must include two additional restraints: the system must be provided with valves capable of closing the primary coolant penetrations of the containment system, and the radioactive material content of the primary coolant must be maintained at sufficiently low levels that it would not cause radiation problems if it were dispersed to the environment under accident conditions. Reliance on such operational equipment for public protection for plants in close proximity to relatively large population centers is, of course, subject to more stringent regulatory restrictions.

Containment requirements for spent fuel handling and storage facilities are dependent upon whether or not decay heat could cause fuel meltdown and subsequent release of gaseous fission products. Wet refueling and fuel storage schemes, where the fuel is continually submerged under a cooling liquid, are usually free from meltdown considerations. Dry refueling operations usually must be conducted within a containment system because of the greater possibility of fuel meltdown in the event of equipment failure. However, since such an accident does not possess the energy to produce high pressures, the type of containment provided for this purpose may be different and may involve separate facilities.

The advantages of limiting the equipment within the containment structure include: (1) reduced personnel exposure to radioactive and potentially dangerous areas, (2) reduced internal shielding and air lock access requirements as a result of reduced personnel access, (3) easy accessibility to equipment remaining outside the containment structure, (4) reduced congestion during construction because the areas for the reactor and turbine facilities are separated, and (5) more flexibility in the construction schedule since the containment vessel and turbine erection programs may be conducted independently.

In contrast, the advantages of placing most of the equipment within the containment structure include: (1) somewhat greater safety to the environs, since more radioactive components are located within the structure, (2) shorter piping runs between various plant components, and (3) elimination of many piping and control penetrations through the containment vessel wall.

Placement of equipment within the containment structure eliminates, in many instances, penetrations for process lines; on the other hand, it may increase the number of control and instrument penetrations for operation of the equipment. Consequently, placement within or without the containment vessel may be considered an advantage or disadvantage depending on the relative leak tightness required and on the effect of placement on the expected reliability of operation of the equipment.

As shown in Table 8.3, all indirect-cycle nuclear power plants built to date in the United States, including the liquid-metal- and gas-cooled plants, have the entire primary coolant loop, including the heat exchangers, enclosed within the containment system. Most U.S. plants have the turbine-generator facilities outside of the containment structure, even though the turbine may be an integral part of the primary coolant loop, as in the case of direct-cycle nuclear plants. The BONUS and EBWR plants are exceptions to this latter point; however, both are developmental facilities.

### 8.1.3 Plant Layout and Structural Considerations

Since the cost of a containment system will be proportional to the quantity and cost of materials and labor required, the size, shape, and materials should be chosen to minimize these items.

Table 8.3. Equipment Items Enclosed Within the Containment Systems of Various Power Reactors

| Reactor Type | Name | Reactor | Heat Exchanger | Turbine-Generator | Fuel Storage |
|---|---|---|---|---|---|
| Indirect-cycle water-cooled reactor | Shippingport | x | x | | |
| | Yankee | x | x | | |
| | Indian Point | x | x | | |
| Direct-cycle water-cooled reactor | EBWR | x | x | x | x |
| | Humboldt Bay | x | (a) | | (b) |
| | BONUS | x | (a) | x | |
| | Big Rock Point | x | (a) | | x |
| Combination direct- and indirect-cycle water-cooled reactor | Dresden | x | x | | |
| Liquid-metal-cooled reactor | Fermi | x | x | | |
| | Hallam | x | x | | (b) |
| Gas-cooled reactor | EGCR | x | x | | |
| | Peach Bottom | x | x | | (b) |

[a]Not applicable.

[b]Fuel storage and shipment operations conducted in separately confined spaces.

#### 8.1.3.1 Size

For a given temperature and quantity of atmosphere assumed for the design accident condition, increasing the net volume of the containment vessel results in a proportional decrease in the design absolute pressure. This will result in a reduced vessel wall thickness, since the pressure decreases with the third power of the linear dimension increase, while the thickness required for a given pressure increases only with the first power of the dimensional increase. However, since the weight of steel required increases both with the wall thickness and with the surface area, which increases with the second power of the linear dimension increase, the total weight of steel remains approximately constant for this idealized case of constant gas quantity.

It has been shown,[6] however, that for most actual cases, the weight of steel required decreases slightly with decreasing volume because, at the higher pressure, vaporization of water is suppressed so that the quantity of gas is not constant and the pressure increase is not directly proportional to the volume decrease. Consequently, it is commonly accepted that the minimum cost for spherical or cylindrical steel vessels will occur at minimum volume, provided that the major cost is associated with procurement of the steel. However, the wall thickness increases with decreasing volume and, for wall thicknesses above certain specified values, the ASME code for steel pressure vessels requires postweld heat treatment, which is an expensive and difficult fabrication procedure when done in the field. For some designs, postwelding heat treatment would be destructive, since the vessel would collapse under its own weight if heated to the required temperature. The minimum total cost point usually occurs with the minimum volume vessel having field-made welds that do not require postwelding heat treatment.

In addition, if containment shielding is to be provided, a reduction in the containment dimensions will also reduce the quantity of shielding material and structural supports required and thereby reduce a related cost item.

A reduction in the containment vessel diameter by the use of multiple vessels usually does not reduce the amount of material required, since the decreased volume per unit length of vessel makes a compensating increase in the number or length of vessels necessary. The additional problems and costs associated with connecting ducts, especially those caused by thermal expansion of the vessels, make such multiple-vessel containment systems relatively unattractive. However, as reactor plant sizes increase, the economic limit of using the maximum wall thicknesses without field postwelding heat treatment may make multiple-vessel containment systems become relatively more attractive. The Shippingport reactor containment system, constructed prior to the bulk of the existing single-vessel containment system experience, is an example of a multiple-vessel system having separate, connected containment vessels for the reactor, for each of two steam generators, and for auxiliary equipment.[7] This system is also discussed in Chapter 7.

Reinforced-concrete containment vessels are not limited in wall thickness by the requirements for field postwelding heat treatment and thus are attractive for large reactor plants because greater loads per unit wall length can be used than may be practical with steel. Concrete vessels are even more advantageous if containment shielding is required, since the containment and shielding functions can be combined (see Sec. 8.2.2).

In the preceding discussion it was assumed that the containment vessel must be designed for the entire energy release from the primary system in the event of an accident. Several alternate methods have been developed to reduce the energy to be contained, either by the use of heat sinks, as in pressure-suppression systems, or by the controlled exhausting of the initial low-level radioactive fluids so that the actual amount of fluid contained is reduced. However, the use of these methods does not alter the factors to be considered in selecting containment size or shape, except that the volume and design pressure may be reduced and additional consideration must be given to locating the heat sinks, any ducting necessary, and the exhaust systems. Of course, if the energy content of the fluids to be contained is sufficiently low that a significant pressure buildup cannot occur, the minimum volume compatible with equipment arrangements and access requirements may be adequate.

#### 8.1.3.2 Shape

Although the shapes of containment systems conceivably could vary almost without limit, practical considerations usually restrict containment system shapes to relatively simple volumetric forms, including flat-walled rectangular prisms, cylinders with either flat or curved-surface ends, spheres, and various combinations of these. Determination of the optimum shape of a containment system requires consideration of the site configuration, the natural arrangements possible for the equipment to be installed within the containment structure, whether or not the system must act as a pressure vessel, and the cost of constructing various geometrical shapes.

Whether or not the system must act as a pressure vessel may be the most significant consideration, since lower cost flat-walled rectangular spaces may be used in low-pressure systems. If rather high quantities of stored energy are released under the postulated accident conditions, a sphere rather than a cylinder may be required to avoid exceeding the maximum allowable wall thickness without field postwelding heat treatment. A spherical shape results in the minimum wall thickness and minimum weight of steel, since the steel theoretically is subjected to equal two-dimensional stresses and thus is utilized most efficiently. On the other hand, a sphere may be more expensive to construct, since the steel plate must be formed with double curvature. Also, adjustment of the vessel volume during design may be more difficult because the horizontal and vertical dimensions cannot be varied independently, and thus it is more difficult to effectively utilize all the contained volume. A cylinder has advantages in these respects, being more easily constructed and offering a shape whose height and diameter may be adjusted independently to suit the plant configuration. A cylinder with curved rather than flat ends allows independent height and diameter control but avoids the stress problems associated with flat-ended structures. The use of a flat-bottomed vertical cylinder with a curved head is sometimes possible, since the weight of the structure tends to help the bottom slab resist the bending moment caused by internal pressure and thus reduces the need for a curved bottom surface.

A design review might suggest that the "wasted" space surrounding major components within the containment vessel be utilized by locating various auxiliary components in this space. Relocating a component may be very worthwhile. However, locating additional components within the same containment dimensions will reduce the net free volume of the containment system and thus increase the design pressure. As a general rule, the decision to contain or not to contain any component should be based on whether or not the component requires containment and on the extent to which accessibility to the component is required. Although inserting low-density equipment into available spaces may cost less than providing additional containment space, the space is never free and is never wasted except in those cases where additional pressure can be withstood without going to the next greater available wall plate thickness. This subject is discussed further in Chapter 7.

When site requirements dictate containment shielding, it is sometimes economical to place the containment structure underground or to combine the containment and shielding functions by the use of reinforced concrete structures. In either case, the complexities of construction are likely to override other shape considerations and suggest the use of cylindrical rather than spherical containment shapes.

As discussed in Section 8.1.1, local soil conditions also play an important role in selecting the shape of containment to be used at a given site.

#### 8.1.3.3 Materials

The materials from which a containment vessel is fabricated must be compatible with the vessel shape and with the intended service. As the design internal pressure of the containment vessel increases, it becomes more compelling to use high-strength materials and containment shapes suitable for pressure vessel construction, such as cylinders or spheres. These considerations are not as important for a low-pressure (less than 5 psig) containment structure where more normal building layout and structural loading practices can be followed. A flat-bottomed cylindrical or prismatic structure, or even an industrial-type steel frame or concrete building with modifications for leak tightness, may provide the most economical containment for low or slightly negative internal pressure designs.

For power reactors having large energy release characteristics under the maximum credible accident conditions, it has been the practice in the United States to design and build containment vessels out of steel according to Section VIII of the ASME Boiler and Pressure Vessel Code and the Nuclear Code Cases.[3] Use of the Unfired Pressure Vessel Code is a legal requirement in many states for vessels with design pressures greater than 15 psig. The recently published Section III of the ASME Code, Rules for Construction of Nuclear Vessels,[2] contains essentially the same provisions for containment vessels as are included in the Unfired Pressure Vessel Code (Section VIII) and the Nuclear Code Cases, except that it applies to vessels with design pressures above 5 psig. Section III is also referenced in the ASA Proposed Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors (see Appendix E).

Because of the low pressure resulting from the maximum postulated accident in a sodium-cooled graphite-moderated reactor, the containment structures for this type of plant have usually been reinforced-concrete boxes with welded steel liners. A concrete structure with a welded steel liner is also used for the pressure-suppression pool for the Humboldt Bay direct-cycle boiling-water reactor. This pool is in the shape of a cylindrical annulus wrapped around the reactor drywell. The cylindrical shape was chosen to facilitate caisson construction. The building code requirements for reinforced concrete[8] were the basis for design for these concrete containment structures. Reinforced concrete with a welded steel liner was also used for the containment vessel for the 16-Mw(e) CVTR at Parr Shoals, South Carolina. In this case, a partially aboveground, vertical cylinder with a flat bottom and a hemispherical top was used.[9]

The choice of materials is discussed in more detail in Section 8.2.

### 8.1.4 Additional Requirements

#### 8.1.4.1 Containment Shielding

Biological shielding is sometimes required to protect the adjacent plant facilities and operating and maintenance personnel or nearby offsite areas from the potential direct radiation caused by a nuclear accident. When such requirements exist, concrete containment structures and underground designs have an advantage over other types of structures in that the containment vessel construction materials or the surrounding soil provide most or all of the required shielding.

Aboveground steel containment vessels can be shielded, but there is a resulting plant cost increase. A concrete dome, such as that used for the Indian Point installation, can be used to provide complete shielding of a containment sphere, in which case it might also be used as a secondary containment boundary. Cylindrical concrete shadow shields can be built around either spheres or cylinders if complete shielding against "skyshine" is not required. In some plants employing aboveground steel containment vessels where potential direct radiation to off-site areas is not a problem, only the control room is shielded to protect the operating personnel against containment radiation or to permit operation of undamaged units adjacent to a nuclear unit that has sustained an accident. The shielding cost is thus minimized. Examples of this type of shielding include the Elk River reactor, the experimental gas-cooled reactor, and the Peach Bottom reactor.

#### 8.1.4.2 Site Layout

The size, shape, elevation, and even type of containment are often dictated, in part, by the site layout and proximity to adjacent structures. At a congested site a small or unusually shaped containment structure may be desirable. For example, at Humboldt Bay, where a nuclear unit was added to an existing conventional thermal power plant complex, a compact arrangement was developed to permit common use of certain conventional facilities. This dictated the plant elevation and the need for shielded containment.

#### 8.1.4.3 Refueling

Where wet refueling is used in conjunction with an external spent-fuel pool, it is desirable to have the surfaces of the refueling water inside the containment system and the water in the spent-fuel pool at the same elevation. Otherwise a lock is required between the refueling pit and the spent fuel pool to maintain the two surfaces at their proper levels while transferring fuel under water between the two areas.

### 8.1.5 Examples of Design Considerations Leading to Various Containment Shapes

The principal U.S. power reactors, the shape of containment system used, and additional design considerations that contributed to the selection of the containment shape and type are listed in Table 8.4. The following paragraphs discuss, as examples, the bases for selecting the containment designs for certain specific plants.

Table 8.4. Reactor Containment Shapes

| Reactor | Containment Envelope Shape | Design Pressure (psig) | Additional Requirements |
|---|---|---|---|
| Dresden | Spherical | 29.5 | Shadow shielding of control room |
| Yankee | Spherical | 31.5 | Shadow shielding of control room |
| Big Rock Point | Spherical | 27 | |
| EBWR | Cylindrical | 15 | |
| VBWR | Cylindrical | 45 | |
| Saxton | Cylindrical | 30 | |
| Indian Point | Spherical | 25 | Full shielding |
| Elk River | Cylindrical | 21 | |
| EBR-II | Cylindrical | 15 | |
| Enrico Fermi | Cylindrical | 32 | |
| Pathfinder | Cylindrical | 78 | |
| EGCR | Cylindrical | 9 | Shadow shielding of control room |
| Humboldt Bay | Cylindrical | | Full shielding |
| | Drywell | 72 | |
| | Suppression chamber | 10 | |
| CVTR | Cylindrical | 23 | |
| ESADA-VESR | Cylindrical | 58 | |
| Peach Bottom | Cylindrical | 8 | Shadow shielding of control room |
| SRE | Flat walled | | Low elevation access to reactor head required for refueling |
| Hallam | Flat walled | 6 | |
| LACBWR | Cylindrical | 52 | |

### 8.1.5.1 Spherical

The design of the Big Rock Point plant illustrates the considerations leading to the choice of a spherical steel containment vessel. This plant has a direct-cycle, forced-circulation, boiling-water reactor with an initial electrical capability of 50 Mw. Steam separation is performed in a separate tank external to the reactor pressure vessel. The plant was built on an undeveloped site on the shore of Lake Michigan. Planning criteria required consideration of only one nuclear unit on the site. Soil conditions consist of a sand-clay mixture with adequate bearing capacity and good slope stability.

Even though the plant is located in the northern U.S. and therefore it was decided that the turbine should be enclosed, no serious consideration was given to including the turbine within the containment structure.

Since no provision was required for shielding additional units and since sufficient exclusion area was provided, shadow shielding was not required. Therefore there was no incentive to consider the use of concrete or underground containment to assist in meeting shielding requirements, and steel construction was the logical choice. Soil conditions were favorable for using a curved surface foundation of this type commonly used with a steel capsule or sphere. A pressure-suppression concept similar to that used at Humboldt Bay was not attractive because of the large space occupied by the reactor, the steam separation drum, and the interconnecting piping. The reactor package could be accommodated in either a capsule or a sphere. Bids were solicited for both, and the choice of a sphere was made on the basis of economics.

### 8.1.5.2 Cylindrical

As indicated by Table 8.4, most containment systems have been cylindrical. Although the Humboldt Bay containment design is quite different from other cylindrical containment designs and cannot be considered typical, a review of the considerations that led to the selection of this design is illustrative.

At Humboldt Bay, a single-cycle natural-circulation 50-Mw(e) boiling-water reactor plant was built alongside two existing oil-fired units. Containment shielding was required to permit continued operation of these other units in the event of an accident in the nuclear unit. This requirement imposed a limitation on the types of containment that could be used. The soil consisted of sand, gravel, and clay, and there was a high ground-water table. The existing units were built on piles.

Originally, an aboveground steel capsule located alongside the existing units was considered for this plant. However, the requirement for shielding coupled with the desire to investigate a pressure-suppression system led to the consideration of an underground design. For a single-cycle natural-circulation boiling-water reactor, the containment vessel (drywell) need enclose only the reactor vessel. This requires only a small cylinder, which is especially suitable for underground construction and pressure suppression. The drywell and suppression pool were incorporated into a flat-ended cylindrical concrete structure sunk as a caisson. A caisson was a logical choice since the soil conditions included a gravel aquifer of high permeability, which was close enough to the ground surface that it would be in contact with the containment wall. Since the reactor vessel was cylindrical, it was natural to make the drywell containment a slightly larger cylinder and to wrap the suppression pool around the drywell to form an annulus. This shape is the most efficient to resist the internal suppression-pool pressure and external pressures due to ground water and earth.

### 8.1.5.3 Flat-Walled Rectangular Prisms

Rectangular prism containment structures usually are employed only where the design pressure is less than approximately 5 in. of $H_2O$. Since flat-walled structures are the simplest to construct, this type of structure is given serious consideration for designs in this pressure range.

It is particularly attractive for plants where wall and roof spans can be relatively short so that wall thicknesses do not have to be large. An example of this type of containment, designed for low positive pressures, is the containment structure for the sodium-graphite reactor at Hallam, Nebraska. This design consists of a number of concrete vaults lined with steel plate. In this case, the containment structure was placed below grade for convenient access to the equipment located on top of the reactor and to minimize the shielding requirements. Access is provided via gasketed shielding plugs held in place by gravity.

Flat-walled low-leakage buildings of relatively conventional building materials are also used.[10] In this case, the building air is kept at a slightly negative pressure with respect to the surrounding atmosphere so that all leakage will be inward. Leakage passes through the containment spaces and is exhausted through appropriate air-cleaning and filtering systems located in the exhaust system. The superstructure at Hallam is an example of this type of building. It is of steel frame with fluted steel roof deck and siding. Joints and laps are caulked and fasteners have neoprene washers. At Humboldt Bay, where the maximum credible accident analysis showed the possibility of melting a fuel element during refueling operations, the refueling building over the reactor has the additional function of a shield for "skyshine." Twelve inches of concrete was required for this purpose. Since no significant pressure rise would occur in this area, a concrete building was chosen with poured-in-place walls and a flat roof of precast, prestressed concrete beams and a poured-in-place topping. Doors have special gaskets to limit leakage.

## 8.2 CHOICE OF MATERIALS

### 8.2.1 Steel

Most containment vessels in the United States have been built of carbon steel plate. The principal reason for the choice of carbon steel over other materials has been cost. Almost none of the alloy or stainless steels and nonferrous metals or alloys have been able to compete. Other materials may, however, have some characteristics superior to those of carbon steels with regard to containment, such as greater resistance to low-temperature brittle fracture and greater corrosion resistance. Even with respect to these properties, properly treated carbon steel has been shown to be completely adequate for the expected service conditions. Design and construction procedures for steel vessels are well known and in most cases are governed by established codes, notably, Section VIII of the ASME Boiler and Pressure Vessel Code[3] and its various Nuclear Code Case interpretations; also, subsection B of Section III of the ASME Code[2] applies specifically to containment vessels and includes essentially the same provisions as Section VIII and the applicable Code Cases. For low-pressure containment vessels, the provisions of API Tentative Standard 620, "Recommended Rules for the Design and Construction of Large Welded Low-Pressure Storage Tanks," have also been used.

Paragraph (1)(b)(3) of Code Case 1272N-5 and Paragraph N-1342 of Section III specify that the maximum shell plate thickness permitted without postwelding heat treatment is 1.5 in. (at the welded joint where plates of differing thickness are joined), if the material is preheated to at least 200°F during welding, or 1.25 in., if the material is not preheated. This has placed a practical limit on shell plate thickness, since it is not economic to postweld heat treat a large vessel in the field. This limitation may have a substantial effect on the design of containment vessels for large nuclear plants. The development of designs involving the use of high-strength steel or reinforced concrete may be required for some applications.

Steel is used for most pressure vessels and for many other structures because of its high tensile strength coupled with its ductility. However, steel exhibits a phenomenon known as low-temperature notch brittleness. This characteristic of ferritic steels has resulted in the catastrophic failure of large steel structures,[11] such as ships,[12-14] bridges, pressure vessels, pipe lines, and storage tanks.[15] This phenomenon has received considerable attention, and numerous tests have been devised to evaluate the tendency of ferritic steels to low-temperature brittleness. In general, these tests define a ductile-to-brittle transition temperature below which the material fails in a brittle manner. However, this transition temperature is not an inherent constant of the material but depends on the test conditions and the specimen geometry. Several of the test methods have been correlated with service failures, and this correlation is the basis of present methods of fracture analysis. The various investigations of brittle fracture of steels and other materials indicate that the possibility of brittle fracture depends on the following parameters:

1. The transition temperature of the metal,
2. The service or test temperature,
3. The stress level in the metal,
4. The presence of a crack or sharp notch causing a stress concentration, and
5. The size of the defect.

A more complete discussion of this phenomenon, the parameters involved, and the analysis of engineering structures is presented in Section 8.5, and an excellent analysis of the "Procedures for Fracture Safe Engineering Design" has been prepared by Pellini and Puzak.[16] The procedures outlined by Pellini and Puzak are fairly widely accepted and are the basis for a number of the requirements in the ASME Code and also in some operating procedures in use in the United States today.[17]

Paragraph (1)(b)(1) of Code Case 1272N-5 and Paragraph N-1211 (a) of Section III of the ASME Code specify that, unless a containment vessel is to be heat treated after welding, the steel plate shall conform to specification SA-300 (ASTM Specification A-300). This specification covers steel plate to be used in welded pressure vessels subject to low temperatures. It requires plates to be furnished in a uniformly heat treated condition by normalizing and requires that samples of the plate and welded areas be impact tested using the Charpy U-notch or keyhole impact test as an indication of the resistance of the material to brittle fracture at low temperatures. Paragraph N-1210 of Section III specifies that, for containment vessels exposed to ambient air temperature, impact tests shall be performed at a temperature not less than 30°F below the lowest service metal temperature. Paragraph N-1211 (a) of Section III requires the use of Charpy V-notch specimens rather than the Charpy U-notch or keyhole impact test specimens. Paragraph N-332 and Table N-332 of Section III give various impact values required for various strength levels in steels.

It is highly unlikely that reactor containment vessels will exhibit what is termed radiation embrittlement, which consists of an increase in the ductile-to-brittle transition temperature because of neutron irradiation. Radiation embrittlement of steel is briefly discussed in Section 8.5.

The types of steel plate most commonly used for containment vessels are ASTM A-201 and A-212, both Grade B and Firebox quality, made to the ASTM A-300 specification. The steel plate is also commonly specified to be aluminum killed and heat treated to fine grain size in order to further reduce notch sensitivity. The A-201 steel has often been favored because of its greater ductility and lower notch sensitivity, even though its allowable stress is less than that of A-212 steel. Forgings and pipe are usually specified to be ASTM A-350 and A-333 steel, respectively. Plate for low-pressure containment vessels may be A-131 steel. Properties of these and some other steels suitable for containment structures are given in Tables 8.5, 8.6, and 8.7. The physical properties of all steels used

Table 8.5. Physical Properties of Carbon Steel

Density: 489 lb/ft$^3$
Thermal conductivity: 26.2 Btu/hr·°F·ft
Specific heat: 0.1 Btu/lb·°F
Poisson's ratio: 0.26

| Temperature Range (°F) | Mean Coefficient of Thermal Expansion (in./in.·°F) |
|---|---|
| | $\times 10^{-6}$ |
| 70–200 | 6.38 |
| –300 | 6.60 |
| –400 | 6.82 |
| –500 | 7.02 |
| –600 | 7.23 |
| –700 | 7.44 |
| –800 | 7.65 |

Table 8.6. Modulus of Elasticity of Carbon Steels

| Temperature (°F) | Modulus of Elasticity ($lb/in.^2$) | |
|---|---|---|
| | ⩽0.30% C | >0.30% C |
| | $\times 10^6$ | $\times 10^6$ |
| 70 | 27.9 | 29.9 |
| 200 | 27.7 | 29.5 |
| 300 | 27.4 | 29.0 |
| 400 | 27.0 | 28.3 |
| 500 | 26.4 | 27.4 |
| 600 | 25.7 | 26.7 |
| 700 | 24.8 | 25.4 |
| 800 | 23.4 | 23.8 |

Table 8.7. Mechanical Properties of Some Containment Steels[a]

| ASTM Spec. No. | Type of Steel | Grade | Use | Minimum Yield Point (psi) | Tensile Strength (psi) | Elongation in 2 in. (%) |
|---|---|---|---|---|---|---|
| A-7-61T | | | Structural | 33,000 | 60,000–72,000 | 24 |
| A-36-63T | | | Structural | 36,000 | 60,000–80,000 | 23 |
| A-131-61 | | ABC | Plate and structural | 32,000 | 58,000–71,000 | 24 |
| A-201-64 | Flange | A | Plate | 30,000 | 55,000–65,000 | 28 |
| | Firebox | | | | | 29 |
| A-201-64 | Flange | B | Plate | 32,000 | 60,000–72,000 | 25 |
| | Firebox | | | | | 26 |
| A-212-64 | Flange | A | Plate | 35,000 | 65,000–77,000 | 23 |
| | Firebox | | | | | 24 |
| A-212-64 | Flange | B | | 38,000 | 70,000–85,000 | 21 |
| | Firebox | | | | | 22 |
| A-283-58 | | C | | 30,000 | 55,000–65,000 | 27 |
| A-283-58 | | D | Structural | 33,000 | 60,000–72,000 | 24 |
| A-285-64 | | C | | 30,000 | 55,000–65,000 | 28 |
| A-333-64 | | 3 | Pipes and tubes | 35,000 | 65,000 | 30 |
| | | 4 | | 35,000 | 60,000 | 30 |
| | | 5 | | 35,000 | 65,000 | 30 |
| A-350-64 | | LF1 | Forgings | 30,000 | 60,000 | 25 |
| | | LF2 | | 36,000 | 70,000 | 22 |
| A-442-64 | | 55 | Plate | 30,000 | 55,000–65,000 | 28 |
| | | 60 | | 32,000 | 60,000–72,000 | 25 |
| ASME code case 1280 | | 65 | Plate | 35,000 | 65,000–77,000 | 24 |
| | | 70 | | 38,000 | 70,000–85,000 | 22 |

[a]From ref. 18.

in the past in containment vessels are similar and are listed without identifying the type of steel. The mechanical properties vary sufficiently that they are listed for each type of steel.

The effects of thermal and pressure expansion of steel containment vessels are minimized by allowing the vessel to expand freely, such as was done for the EGCR, Dresden, and many other containment vessels. This requires that all structures connected to access openings and penetrations must permit this movement. This subject is treated in more detail in Section 8.3.

Insulation is often used on steel vessels, both to maintain the plate temperature above the ambient temperature to minimize the possibility of brittle fracture and also to reduce thermal stresses (see Sec. 8.6). On the other hand, insulation will impede heat transport and thereby possibly cause a slightly higher peak pressure and much more prolonged pressure decay in the event of a loss of coolant accident. Corrosion protection is also normally provided and usually includes paint, other coatings, and sometimes cathodic protection (see Sec. 8.6).

### 8.2.2 Concrete

Concrete is a relatively low-cost construction material and is a logical choice for containment structures similar to conventional buildings where the design pressure is low and where a high degree of leak tightness is not required. Concrete also has several important advantages over steel for other types of containment structures. As a result of the greater wall thicknesses required, a concrete structure is less subject to buckling from external loads, such as wind and soil pressure, than is a thin steel plate structure. Further, concrete structures are not susceptible to corrosion when properly constructed and are therefore particularly suitable for underground applications.

If a containment radiation shield is required to protect nearby areas from direct radiation in the event of an accident, it may be advantageous to provide both the shielding function and the containment function with a single concrete structure. A concrete containment vessel was used on the CVTR for this reason, and similar structures are planned for use on the proposed Connecticut Yankee and Malibu plants.

Concrete vessels are not subject to the practical limitations in wall thickness imposed on steel vessels because of the requirement for postwelding heat treatment. Thus it may be possible to build large concrete containment vessels for somewhat higher pressure service than is presently attainable with steel vessels. For some of the large nuclear plants now contemplated, concrete vessels or high-strength steel vessels will possibly provide the only practical means of containing the large amounts of stored energy in the coolant without using pressure suppression or other means of energy removal.

A less tangible advantage that could result from using concrete for containment is greater freedom in reactor plant arrangement. At present, the primary system components of most reactor plants are compactly arranged near the center of the containment structure to minimize the amount of primary shielding material required. If a concrete containment structure is used, it may prove advantageous to locate some components near the containment wall and utilize this wall for primary shielding, as well as for containment and for containment shielding.

There are several limitations to the use of concrete for containment vessels. Since concrete has little tensile strength of its own, steel reinforcing is nearly always required, particularly in an application such as a pressure vessel where the loading is primarily in tension. Since a reinforcing bar can take loads in one direction only, whereas steel plate can be loaded biaxially, more steel may be required for a reinforced concrete vessel than for a comparable all-steel vessel. However, since reinforcing bars have a lower unit cost than plate, the greater steel weight for concrete construction is partially offset costwise. Since concrete is subject to cracking, a liner is required to provide a degree of leak tightness comparable with that obtainable with a steel containment vessel. Therefore, a concrete vessel may cost more than an equivalent steel vessel for those vessel sizes and design pressures for which steel can be used without postwelding heat treatment. If containment shielding is required, however, as in the case of the CVTR, the concrete vessel may be the more economical.

In general, temperature gradients present a greater problem in concrete than in steel, since an internal temperature rise causes thermal expansion that places the outer face of the wall in tension. Since the tensile strength of concrete is much less than that of steel, cracking of the surfaces may occur if the difference in expansion between the internal and external surfaces is too great. The lower thermal conductivity of concrete and the normally thicker walls compound this problem. The normal daily rise and fall of the temperature is often rapid enough that thermal equilibrium is not reached, in which case thermal expansion may not be a problem; however, if necessary, the exterior surface can be insulated.

Ordinary concrete is normally used in containment structures and is no different from concrete used in other structural applications. Concrete used for shielding, however, requires special care in design and placement to prevent voids or cracks that may affect shielding effectiveness. High-density concrete may be used for shielding when limited space is available. Since a premium cost is associated with high-density concrete, its use is limited to special conditions. Concrete for a containment barrier must have low porosity and must be free of gross cracks to minimize leakage of air. However, since it is difficult to ensure that a large concrete structure will be crack-free, a coating or steel liner is usually required to assure low leakage.

Various properties of several concrete mixes used in containment structures and shields are listed in Table 8.8. These are typical values used in engineering design and analysis and were compiled from various sources. Unless otherwise indicated, the values given are for ordinary concrete.

Table 8.8. Some Properties of Concrete

| | |
|---|---|
| Density, $\rho$, $lb/ft^3$ | |
| Ordinary concrete[a] | 145 |
| High-density concrete[b] | |
| Limonite concrete | 185 |
| Limonite-magnetite concrete | 215–225 |
| Barytes concrete | 220–225 |
| Magnetite concrete | 210–245 |
| Ferrophosphorous concrete | 300 |
| Iron-portland concrete (steel shot and steel punchings used as aggregate) | 360–410 |
| Compressive strength at 28 days,[c] $f_c'$, psi | |
| 4 gal of water per sack of cement | 6,000 |
| 5 gal of water per sack of cement | 5,000 |
| 6 gal of water per sack of cement | 4,000 |
| 7 gal of water per sack of cement | 3,300 |
| 8 gal of water per sack of cement | 2,800 |
| Tensile strength[d] | 7 to 11% of $f_c'$ |
| Modulus of elasticity at 28 days,[a] $E_c$ | |
| For densities, $\rho$, between 90 and 155 $lb/ft^3$ | $33\ \rho^{1.5}\ (f_c')^{1/2}$ |
| For ordinary concrete | $5.8 \times 10^4\ (f_c')^{1/2}$ |
| Poisson's ratio[e] | 0.15 to 0.24 (0.20 average) |
| Coefficient of thermal expansion,[e] $\alpha$, in./in.·°F | |
| Range | $4 \times 10^{-6}$ to $7 \times 10^{-6}$ |
| Average | $6.5 \times 10^{-6}$ |
| Thermal conductivity,[f] k, Btu/hr·ft·°F | |
| Ordinary concrete | 0.2 to 1.0 |
| Iron ore aggregate concrete | 1.2 to 3 |
| Specific heat,[e] $c_p$, Btu/lb·°F | 0.20 to 0.24 |
| Permeability (air),[g] cfm per inch of thickness per $ft^2$ per $\Delta P$ (in. $H_2O$) | $2 \times 10^{-5}$ to $3 \times 10^{-6}$ |

[a]From ref. 8.
[b]From refs. 19 and 20.
[c]From ref. 21.
[d]From ref. 22.
[e]From ref. 23.
[f]From ref. 19.
[g]From ref. 10.

### 8.2.2.1 Reinforced Concrete

The term reinforced concrete as used in this chapter applies to concrete that contains mild-steel bars for reinforcing. Prestressed concrete containing high-strength steel stressing cables or bars is discussed later in this section.

The reinforced-concrete CVTR containment vessel at Parr Shoals, South Carolina, was built under the provisions of the Building Code Requirements for Reinforced Concrete.[8] This code covers the design of reinforced-concrete buildings, but its provisions can be applied, with modifications, as necessary, to other specialized reinforced-concrete structures. The Building Code permits considerably higher stresses in the reinforcing steel than the corresponding basic plate stress allowed by Section VIII of the ASME Code.

The shapes used for reinforced-concrete containment vessels may differ from those used for steel vessels because the different properties of the materials create different problems. Steel reinforcing bar patterns can be easily worked out for cylinders, but the design of a roof for a cylinder is more difficult. A hemispherical dome theoretically makes most efficient use of the material, but because the meridian reinforcing bars converge and thus the constant optimum spacing cannot be maintained, the theoretical efficiency cannot be realized. On the other hand, a flat roof requires considerable thickness to overcome large bending stresses, and unless the diameter is quite small, it is uneconomical. Consequently, a hemisphere or other curved surface is usually used.

Penetrations in concrete structures can be similar to those in steel structures. Normally they are made through conduits placed in the structure as forms prior to placement of the concrete. The conduit is sealed into the containment envelope boundary by welding, if a metal liner is used, or by applying a coating to the mating surfaces, if a non-metallic coating is used to seal the concrete. Sealing of the penetrating components inside the conduits is accomplished by the same techniques as those used for steel vessels, which are outlined in further detail in Chapter 9. Structural problems are negligible if the design pressure is low or the diameter of the penetration is small. However, large penetrations designed to operate in high-pressure systems must be structurally tied into the steel reinforcing bar, and the reinforcing bar that is interrupted by the penetration must be properly anchored to a structural steel ring or frame. If the reinforcing bar is fastened to the ring by welding, it must be done under carefully controlled conditions because of the high carbon content of the reinforcing bar.

Either a steel liner or a nonmetallic coating may be used as a membrane to reduce leakage from concrete containment vessels to acceptable levels. A steel membrane has high reliability and can provide the same integrity as an all-steel containment vessel if it is properly constructed and leak tested. However, differential thermal expansion may tend to cause an inside liner to buckle. This in itself may not be serious, but there is increased possibility of rupture of the lining. If somewhat higher leakage is permissible, a nonmetallic membrane of the type installed at HWCTR or BONUS may be used. However, locating and repairing a defect in a nonmetallic membrane is difficult. (Membranes are discussed more fully in Section 8.4.3.)

The construction problems associated with reinforced concrete structures are different from those of steel containment vessels. More field work is required, and this may result in a longer construction schedule and additional problems in quality control. Many problems are associated with forming and placing a long-span concrete roof, whether flat or dome shaped. The problem of splicing reinforcing bars is difficult, even if the bars are lapped, welded, or connected with mechanical connectors, particularly if there are three or more curtains of bars. If the concrete containment wall is to serve the additional function of a radiation shield, care must be taken to assure uniform distribution of the aggregate, particularly if heavy aggregates and steel punchings are used.

Current practice is to use heavy high-strength steel reinforcing bars in the design of large structural members. Bars with a cross-sectional area of 4 in.$^2$ and a tensile strength of 100,000 psi are now available and are frequently used in general construction practice.

Splicing of reinforcing bars may be done by overlapping, by the use of mechanical connectors, or by welding. Overlapping by the normal 30 bar diameters is the easiest and most common method of construction splicing, but the use of bars of increasingly larger cross-sectional dimensions tends to make this method more difficult and to make splicing with mechanical or welded connections more economical.

Mechanical connectors may be divided into two classes: those that grip prepared surfaces of the ends of the bars with mechanical teeth, as shown in Fig. 8.12, and those that grip the ends through an alloy metal that is allowed to solidify between a grooved connecting sleeve and the normal bar surface deformations, as shown in Fig. 8.13. Mechanical gripping connectors allow the use of the full physical properties of the bars provided the ends are properly smoothed so that the gripping teeth can be properly seated and provided the ends are smoothly cut and butted together so that the coupling does not loosen with working. Alloy-metal mechanical connectors require less bar end preparation but more field construction time to assemble the melted alloy preparation equipment and to make the splice. Metal-alloy butt-spliced mechanical connectors are capable of developing the ultimate strength of the reinforcing bar in compression and the yield strength in tension if the alloy metal and connecting sleeve are properly selected. Less costly grades of alloy metal are often used when full strength splices are not required.

Welding of reinforcing bars has only recently gained wide acceptance. Recommended practices for such welding have been approved by the American Welding Society (AWS D12.1-61) and have been published, along with a descriptive article, by Amirikian.[24] The reinforcing bar normally used in the building industry comes in three grades: structural, intermediate, and hard. The structural grade has chemistry comparable with that of mild structural steel and presents no special problems in welding, but the intermediate and hard grades, mainly because of their higher carbon content, require special procedures to assure an acceptable quality of weld. For bars of intermediate grade (up to 0.50% carbon), low-hydrogen electrodes and preheat and interpass temperature control ranging from 100 to 400°F may be needed. For the hard grade, in which the carbon content varies from 0.50 to 0.80%, the use of Thermit welding or pressurized-gas welding is recommended. Some limitation on the manganese content of the steel may also be necessary. Accordingly, since chemical composition is normally

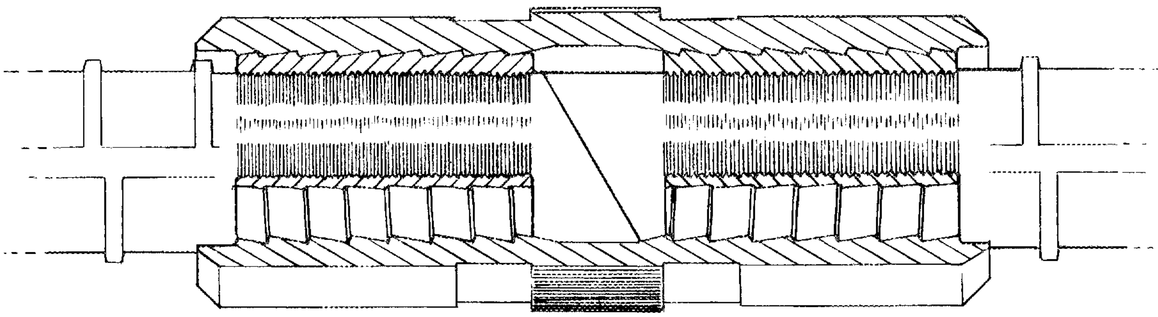

Fig. 8.12. Reinforcing-Bar Mechanical Connector with Teeth Gripper.

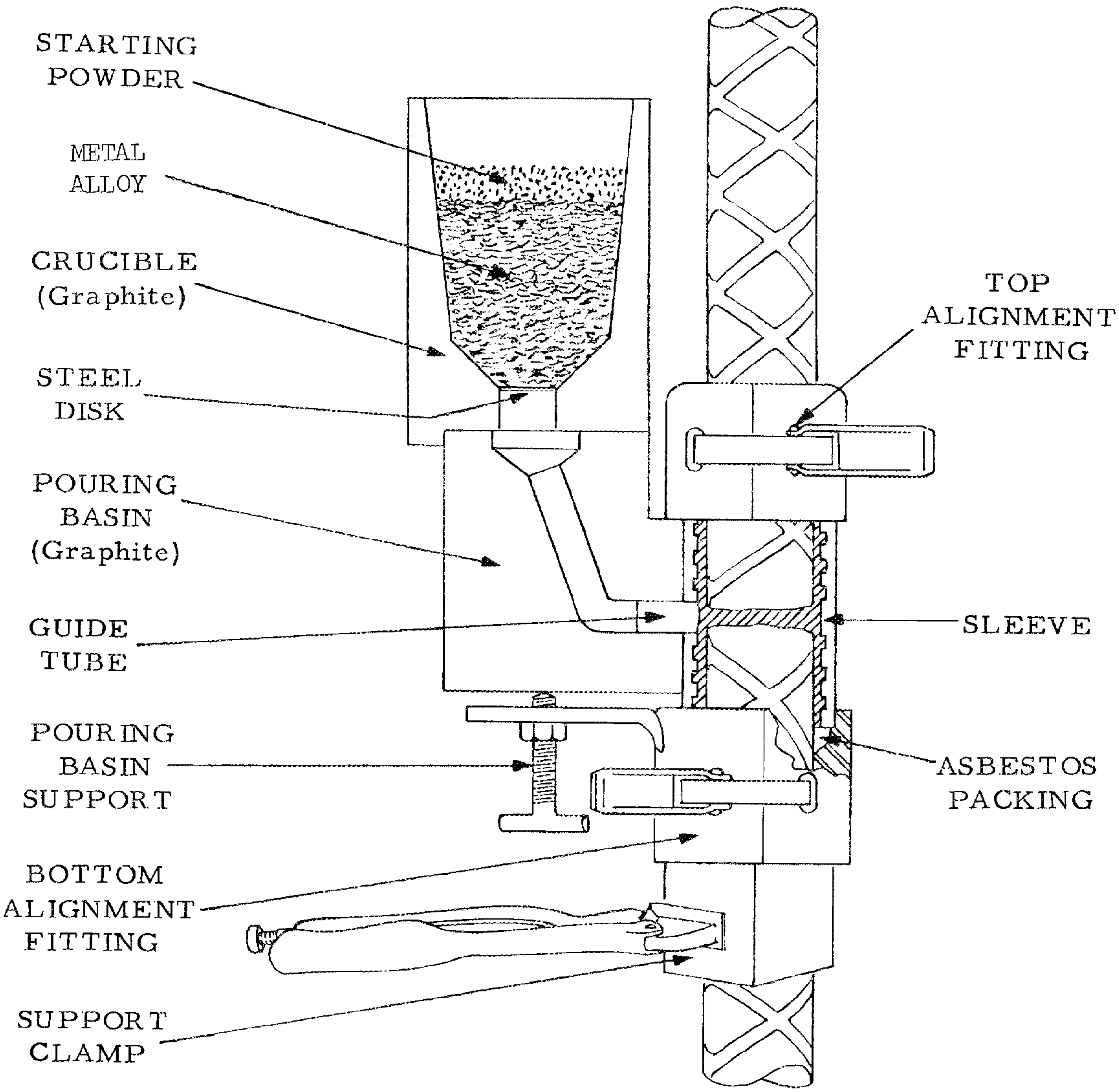


Fig. 8.13. Reinforcing-Bar Mechanical Connector with Alloy-Metal Gripper.

not specified in procuring reinforcing bar, it is important that information regarding the chemical composition be obtained from the supplier in order to determine the proper welding procedures.

The allowable unit stresses for butt welds are taken as equal to those of the base metal or the bar. The strength of a fillet weld made on the ends of splicing sleeves is assumed equal to 68% of the allowable tensile stress for the base metal, bar, or sleeve. Allowable stresses in lap welds with adjacent reinforcing bars or with flat surfaces are reduced because the geometry of the weld is not favorable for obtaining a full penetration or fusion through the root and because the weld is subject to shear rather than tensile or compressive forces. Typical values for mechanical properties of steel reinforcing bars are listed in Table 8.9.

Table 8.9. Properties of Steel Reinforcing Bars[a]

| ASTM Spec. No. | Type of Steel | Grade | Minimum Yield Point (psi) | Tensile Strength (psi) | Elongation in 8 in. (varies with bar size) (%) |
|---|---|---|---|---|---|
| A-15-64 | Billet steel | Structural | 33,000 | 55,000–75,000 | 11–16 |
| | | Intermediate | 40,000 | 70,000–90,000 | 7–11 |
| | | Hard | 50,000 | 80,000 (min) | (b) |
| A-408-64 | Billet steel | Structural | | 55,000–75,000 | 13 |
| | | Intermediate | | 70,000–90,000 | 10 |
| | | Hard | | 80,000 (min) | 7 |
| A-432-64 | Billet steel | | 60,000 | 90,000 | 7 (min) |
| A-431-64 | High-strength billet steel | | 75,000 | 100,000 | 5–7 1/2 |
| A-160-64 | Axle steel | Structural | 33,000 | 55,000–75,000 | 16 (min) |
| | | Intermediate | 40,000 | 70,000–90,000 | 12 (min) |
| | | Hard | 50,000 | 80,000 (min) | (c) |

[a]From ref. 18.

[b]1,000,000 divided by the tensile strength less 1 to 5%.

[c]1,000,000 divided by the tensile strength.

### 8.2.2.2 Prestressed Concrete

Prestressed concrete has many of the same advantages and disadvantages as reinforced concrete. One principal additional advantage is that the concrete is always in compression so that cracking is minimized and leakage through the concrete is reduced. Also, since prestressing wires and rods can be used at higher stress levels, less steel is required. Unless the steel is grouted in position, individual wires may be repeatedly tested, retensioned, or even replaced.

To date, no completely prestressed concrete containment vessel has been built in the U.S., the closest approach being the HWCTR,[25] which has a steel dome and a prestressed concrete cylinder substructure. Foreign experience with large prestressed concrete pressure vessels has been more extensive. Prestressed concrete was used for the large reactor vessels for the gas-cooled G-2, G-3, and EDF-3 reactors in France[26] and for the Oldbury reactors in England.[27] The basic principles of these designs are shown in Figs. 8.14, 8.15, and 8.16, and pertinent data are presented in Table 8.10. Although designed for higher pressures than are likely to be required of containment vessels, the principles of their design and construction may be useful in containment system design. Further discussion of these reactor vessels has been presented by Bender.[28]

Prestressing can be applied most easily to plane and cylindrical surfaces. The problems associated with prestressing spherical shapes are much more difficult because of the biaxial nature of the stresses induced.

Penetrations in prestressed concrete require the same treatment as those in reinforced concrete. The stressing cables can be guided around small penetrations. Larger penetrations must have frames in which the stressing cables are anchored and which distribute the stresses. The Oldbury reactor vessels in England have penetrations for the blowers that are 8 ft 6 in. in diameter.[27]

Steel wire that is not bonded to the concrete should normally be protected against corrosion by galvanizing or by coating with tar, oil, or other waterproof material. Recent catastrophic failure of a prestressed

ORNL Dwg. 63-2792

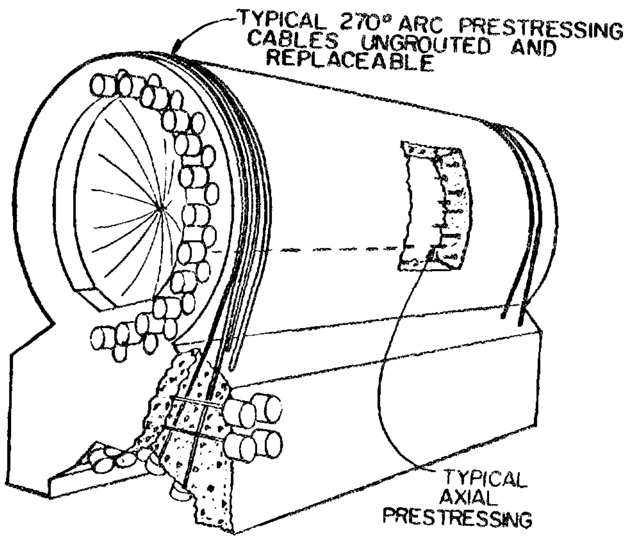


Fig. 8.14. Marcoule G-2 and G-3 Type of Cylindrical Prestressed Concrete Pressure Vessel. (From ref. 28)

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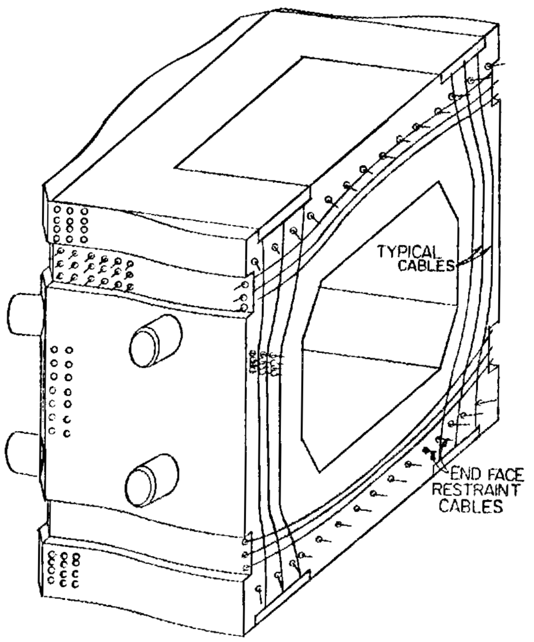


Fig. 8.15. EDF-3 Culvert Concept. (From ref. 28)

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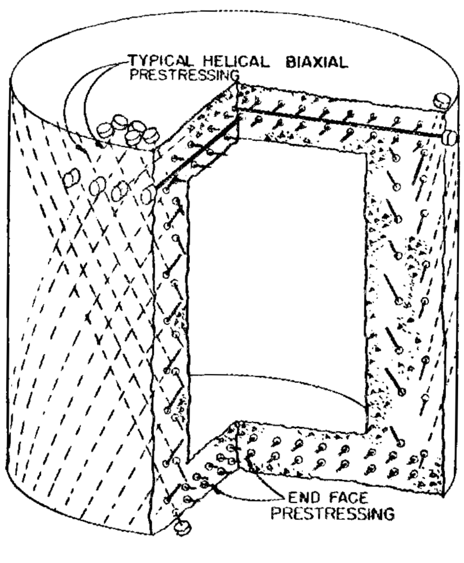



Fig. 8.16. Oldbury 60° Helical Wrap Concept. (From ref. 28)

Table 8.10. Pertinent Data on Prestressed-Concrete Vessel Designs[a]

| | Reactors G-2 and G-3 at Marcoule | EDF-3 Reactor | Oldbury Reactor |
|---|---|---|---|
| Shape | Cylindrical with concave heads | Cubical | Cylindrical with flat ends |
| Prestressing method | 270° wrap, anchored to foundations | Culvert arrangement | 60° helical wrap, horizontal layers in ends |
| Design pressure, psig | 200 | 386 | 350 |
| Prestressed-concrete thickness, ft | 10 | 16 | 15 (walls) 22 (ends) |
| Inside vessel diameter, ft | 46 | 62.5 | 77 |
| Cable size | 797 strands of 5-mm wire | 61 strands of 1.5-in.-diam single cable | 12 groups of 1/2-in.-diam seven-strand cable |
| Cable tensile strength, psi | ~200,000 | ~250,000 | ~250,000 |
| Vessel liner thickness, in. | 1 | 1 | 1/2 |
| Liner cooling method | $CO_2$ at 25°C | Reflective insulation with water coils attached to liner | Reflective insulation with water coils attached to liner |
| Concrete temperature limit, °C | 70 | 70 | 70 |

[a]From ref. 28.

concrete sludge tank as a result of corrosion of the prestressing wire illustrates the need for careful attention to this phase of design.[29] The steel wires may also be drawn through protective guide tubes, which are usually bonded to the concrete by embedding the empty tubes in the forms prior to concrete placement. The wires may be bonded to the concrete by original placement or by subsequent grouting.[30]

Even though the possibility of cracking is lower for prestressed concrete than for reinforced concrete, an inner liner is normally required to achieve leakage rates comparable with those of steel containment vessels. A thermosetting plastic liner is used on the concrete portion of the HWCTR containment structure.

Techniques for construction of a prestressed-concrete containment vessel are not much different from those for reinforced concrete, except that there are additional operations connected with setting and stressing the prestressing elements that require greater skill and involve increased labor costs. Access must be maintained to the cable anchorages until prestressing operations are complete. Thus, portions of the tensioning operations must wait until all major equipment is set in order that large construction openings can be closed. (For further, more detailed information on the theory and design practices for prestressed concrete, see refs. 31 and 32 and the Journal of the Prestressed Concrete Institute.)

### 8.2.3 Combination Steel and Concrete

In a few cases, steel plate and concrete have been used in combination to realize the advantages of each and to obtain an economical containment structure. Concrete is a logical choice for the portions below grade, since it can serve as the foundation and can resist soil and water pressures. A flat-bottomed concrete substructure, as used both at HWCTR[25] and BONUS,[33] also has layout and construction advantages. The corrosion resistance of concrete is particularly beneficial in inaccessible areas where corrosion is difficult to detect and to correct. A steel superstructure, on the other hand, is an efficient means for containing internal pressure in an area where there are few other requirements.

A disadvantage of combination structures is that the overall leakage rate will be determined primarily by leakage through the concrete portion or through the concrete-steel joint, so the full benefit of a steel vessel cannot be realized. Leakage of the HWCTR containment structure, which is of a composite design, was over 0.5% in 24 hr, whereas 0.1% is easily obtainable for all-steel vessels (see also Chap. 7). There is also a structural problem at the junction of the steel plate and the concrete, particularly for high-pressure containment vessels, since the differential circumferential strain must be taken up by a flexible joint or by flexing of the steel vessel wall. On the HWCTR containment vessel, designed for a pressure of 24 psig, over 300 prestressed anchor bolts 3 ft long were used at the junction of the concrete and steel,[25] as shown in Fig. 8.17. On the 5-psig BONUS containment vessel, holddown was accomplished with steel bars on 12-in. centers.[33] Thermal stress problems in the two portions are not altered by the combination construction except at the junction.

Construction of the composite structure is somewhat more complicated than for either steel plate or concrete, since all steps unique to each type enter into combination construction. Two-stage construction (see Sec. 8.4.2) was used at HWCTR, where the internal concrete was completed before steel erection was begun. Two-stage construction was not possible at BONUS, since it was desired to leak test the bottom membrane before covering it with interior concrete. This leak test required completion of the steel dome prior to construction of the contained plant structure.

### 8.2.4 Low-Leakage-Rate Conventional Buildings

The leakage rate of a building is dependent on the basic construction materials used and the care taken to block all discontinuity openings through the use of various filler substances. When a conventional building is used as a containment structure, a reduced-pressure system is usually employed so that an in-leakage rate of several percent per day will be acceptable. The exhaust air is ducted through filters.

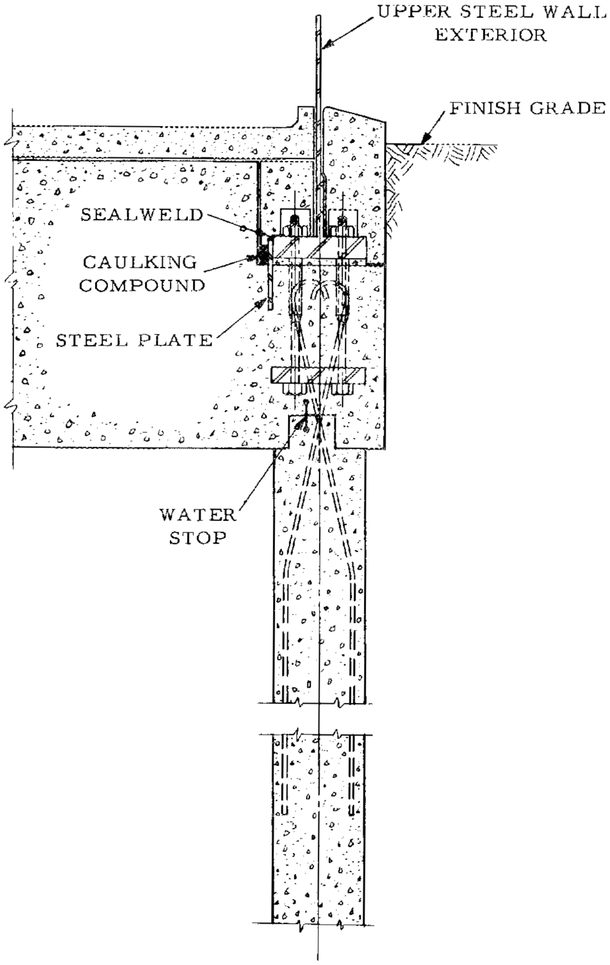



Fig. 8.17. HWCTR Concrete Wall and Steel Joint at Grade. (From ref. 25, Exhibit E)

Extensive tests have been made of the air leakage rates through structural components of metal panel and concrete buildings that might serve as housings for nuclear reactors.[10] These tests have shown that normally constructed buildings can be expected to leak on the order of 2400% of their contained volume per day. However, with improved construction techniques, such as the use of caulked joints and rubber gaskets on metal-screw connectors, the leakage rates of large metal-panel buildings can possibly be reduced to 1000% of the contained volume per day at pressure differentials less than 0.5 psi or 100% per day at 0.1 psi, provided the structure is sufficiently stiffened so that pressure differential, wind, or thermally induced deflections do not open up caulked joints in the metal siding. Based on his experiments, Koontz[10] estimates that large concrete buildings can be constructed that have a leakage rate of less than 1% per day with a pressure differential up to 0.5 psi.

Koontz has developed leakage rate information on various building components for confinement and low-pressure containment buildings. He reports that the accuracy of using component test data to estimate the leakage of a large structure is about 50%. Additional details on penetrations and seals are given in Chapter 9.

## 8.3 SPECIAL DESIGN FEATURES

### 8.3.1 Containment Foundation and Support

The foundation and support requirements of containment systems are dependent on the soil and seismic conditions, the size and weight of the containment system and its contained equipment, and the elevation of the system with respect to ground level. The discussion of foundations and soil mechanics that follows is not intended to present design criteria but rather to introduce some of the important facets that relate to containment structures.

The bearing pressures that may be allowed on soil vary over a wide range, making it mandatory to determine the nature of the underlying soil by borings, test pits, or other methods. If the soil consists of medium or soft clay, a settlement analysis based on consolidation tests of undisturbed soil samples from the foundation strata is needed.

Table 8.11 gives the general classifications of soil and the pressures which they may safely support.[34] The safe values approximate the pressures allowed by most building codes. Higher values may be used if substantiated by specific tests. Conservative values should be used for foundations for stacks or other tall structures because of the magnified effects that would result from a small local settlement.

Footings or pilings and foundations enable the concentrated loads of the containment system to be spread over a sufficient area or to a sufficient depth that the unit pressure will come within the allowable limits. Footings are normally of concrete and often are reinforced with steel to provide better structural distribution of the heavy reactor vessel and shielding loads. Pilings are used where satisfactory bearing soil is not available at a reasonable depth. Materials commonly used are wood, which may carry 8 to 12 tons per pile; concrete, carrying 25 to 60 tons per pile; and structural steel H columns, with still higher capacities. Texts by Abbett[34] and Tschebotarioff[35] or similar engineering documents should be consulted for additional details on the normal design and construction practices for these items.

The load the foundation must carry will vary significantly with the type of containment and the design approach, as well as with the physical size and weight of the nuclear plant contained. The weight of shielding material may easily be the dominant contributor to the total weight and could vary widely depending on plant arrangement, shielding criteria, etc. The buoyant forces acting on underground portions may counteract a significant portion of the total load. For example, the cylindrical, underground, pressure-suppression containment system used at Humboldt Bay has a total weight of 13,080 tons or approximately 5 tons/ft$^2$. However, the buoyant force of 5130 tons reduces the total pressure on the soil to

Table 8.11. Bearing Values of Soils Considered as Supports for Foundations[a]

| Material | Allowable Bearing Value ($tons/ft^2$) |
|---|---|
| Massive crystalline bedrock, such as granite, diorite, gneiss, and traprock in sound condition | 100 |
| Foliated rocks, such as schist and slate, in sound condition | 40 |
| Sedimentary rocks, such as hard shales, siltstones, limestones, sandstones; also thoroughly cemented conglomerates in sound condition | 15 |
| Soft or broken bedrocks of any kind, except shale | 10 |
| Exceptionally compacted or partially cemented gravels, sands, and hardpan | 10 |
| Gravel, sand-gravel mixtures, compact | 6 |
| Gravel, loose; coarse sand, compact | 4 |
| Coarse sand, loose; sand-gravel mixtures, loose; fine sand, compact | 3 |
| Fine sand, loose | 2 |
| Stiff clay and soft shales | 4 |
| Medium-stiff clay | 2 1/2 |
| Medium-soft clay | 1 1/2 |
| Fill, silt, organic material, muck, peak, etc. | 0 |

[a]From ref. 34.

2.85 $tons/ft^2$. In this case, the 60-ft-diam flat bottom of the concrete caisson is sufficiently rigid and in contact with sufficiently firm soil that no further foundation is required. For the Big Rock Point plant, whose 16,000-ton spherical containment vessel is primarily above grade, the pressure on the soil is only approximately 2.3 $tons/ft^2$ because of the 7600 $ft^2$ of foundation bearing area available. This pressure includes about a 0.2-$ton/ft^2$ loading caused by the foundation itself. In this case, the containment vessel foundation extends only to the boundary at which the spherical vessel emerges from the finished grade, but considerable foundation strength was required because the maximum reactor and shielding-load pressures occur in the center of the foundation upper surface.

Normal construction procedures for steel containment vessels call for erection of free-standing plates, for which access to both sides of all welds is possible, prior to the placement of the portion of the foundation that forms the permanent direct support of the vessel. This procedure is used to facilitate welding and to permit full inspection of the welds. It also avoids the difficulties that might be encountered if the foundation were poured first and the steel vessel subsequently fitted to its surface.

Free-standing steel containment vessels must be temporarily supported during construction prior to the placement of external or internal concrete foundations adjacent to the skin surfaces. For spherical vessels, this support has been typically supplied by a set of external columns and bracing that support the vessel from a belt or intermediate point on its girth. Since the total weight to be supported is only the containment shell and related test equipment, i.e., much less than the total operating weight, these support columns will normally require only spread footings or pile caps, if piles are used in the permanent foundation system. Bracing is required in order to resist lateral seismic or wind loads on the vessel. These temporary support columns and bracing may be removed after the foundation concrete is placed and set, as was done at the Fermi plant,[36] or they may be left in place after readjustment as permanent supports or after being loosened so that they do not subsequently induce extraneous stresses in the free-standing vessel.

The placement of concrete under a free-standing steel vessel must be done in steps or by making alternate placements of concrete inside and outside the vessel wall (see Figs. 8.27 and 8.28 of Sec. 8.4.2) so as to eliminate the extensive uplift that could occur as a result of wet concrete fluid pressure. As discussed in Section 8.3.3, steel containment vessels require careful design of the area at which the shell plate leaves the concrete foundation, since this area is subjected to complicated stress patterns, both under normal operating thermal variations and under design pressure and thermal expansion conditions.

### 8.3.2 Penetrations in Containment Vessels

Penetrations in a pressure vessel interrupt the stress pattern and require adequate design followed by careful installation in order to avoid being the weakest link in the vessel. In addition, they provide potential leakage paths and often present difficult sealing problems. This section discusses only the structural aspects of containment vessel penetrations, irrespective of function. The various types of penetrations used for containment vessel access, piping, and electrical connections and the methods of sealing for leak tightness are discussed in Chapter 9.

Rules for the design of small penetrations (under 40 in. in diameter) in steel containment vessels are well covered in Section VIII of the ASME Code. Rules for larger openings and their spacing are given in the proposed ASA standard on design, fabrication, and maintenance of steel containment structures (see App. E). Paragraph N-1342(b) of Section III of the ASME Code specifies that all plates containing penetrations, regardless of size, must be furnace heat treated before butt welding into the containment shell unless the entire vessel is heat treated after completion. Where penetrations would intersect a shell seam or where the penetrations are in a group, a thick insert plate assembly (greater than 1.5 in.) may be used to include one or all of the penetrations, but the periphery of the heat-treated insert assembly must be tapered to keep the thickness of the weld to the adjacent shell plates within the 1.5-in. maximum. The proposed ASA standard recommends that the insert welds be located so as to miss the main butt welds by 12 in. or more, in some cases, or shall be made to cross the main seam at an angle of not less than 30°. Paragraph N-1350 of Section III requires full radiography of Category A and B welded joints, and also of Category C and D joints if a radiographable joint is used. If not radiographable, Category C and D joints, and all other joints not included in Categories A, B, C, and D, shall be examined by ultrasonic, magnetic particle, or liquid penetrant techniques. The joint categories are described in Paragraph UW-3 of Section VIII.

Penetration plates too large to ship in one piece should be assembled and heat treated in the field. According to ASME Code Case 1272N-5, opening frames must be heat treated before being welded into the shell. If a door frame or gasket is included in a large penetration plate, the dimensional limits of available machining equipment may determine the maximum size of such a penetration. Frequently, reinforcing rings for penetrations of these large sizes are forgings made to specification ASTM-A350.

Accepted rules governing the design, location, and reinforcement of penetration frames in concrete containment structures are not available. In general the penetration sleeves are steel for convenience in making attachments. Annular rings are added to the outside to provide anchorage and a labyrinth seal. Penetrations should be arranged in vertical and horizontal rows to provide ligaments for reinforcing bars and to facilitate placement of concrete. Large penetrations will probably require structural steel rings or frames to which reinforcing bars are welded. These rings should be heat treated after welding on the reinforcing bar stubs.

### 8.3.3 Thermal and Other Motions of Containment Vessels

Various thermal and load conditions cause movement of containment vessels and their related fixed accessories, such as piping penetrations and airlocks. These motions can be surprisingly large, amounting to several inches in the case of large steel containment vessels, and under the influence of the sun, the movement of exposed vessels is complex and variable. It is not desirable to introduce restraint of these motions above the foundation; hence, interior and exterior structures should be kept well clear of the containment vessel wall.

If restraints are unavoidable, special provisions must be made where the steel shell emerges from the restraint, such as at the point where a sphere or a cylinder emerges from the concrete foundation. Above the concrete, radial distortions due to temperature and pressure occur, whereas below this point they are suppressed, and high local stresses can be produced. In order to minimize these stress concentrations, a transition zone is normally provided to permit gradually diminishing rigidity of the foundation restraint. An example of one design solution is shown in Fig. 8.18, where a sand-filled tapered space is provided between the outside concrete cradle and the steel shell to furnish a region of intermediate restraint and prevent severe bending when the vessel is pressurized.[37] This area is further protected by a vertical skirt that creates a dead air space and minimizes temperature variations caused by varying atmospheric conditions. A similar detail for a steel cylinder is shown in Fig. 8.19.

Relative motion of the steel plate with respect to the concrete when the containment vessel is pressurized must also be considered. Although opinions differ as to whether this slippage actually occurs, it is good practice to assume that some does occur and to wrap all projections of the steel shell that are to be embedded in the concrete with compressible material. It is also good practice to keep the center of gravity of the interior structure as close as possible to the vertical axis of a spherical containment vessel to avoid the possibility of rotational slippage due to eccentric loading.

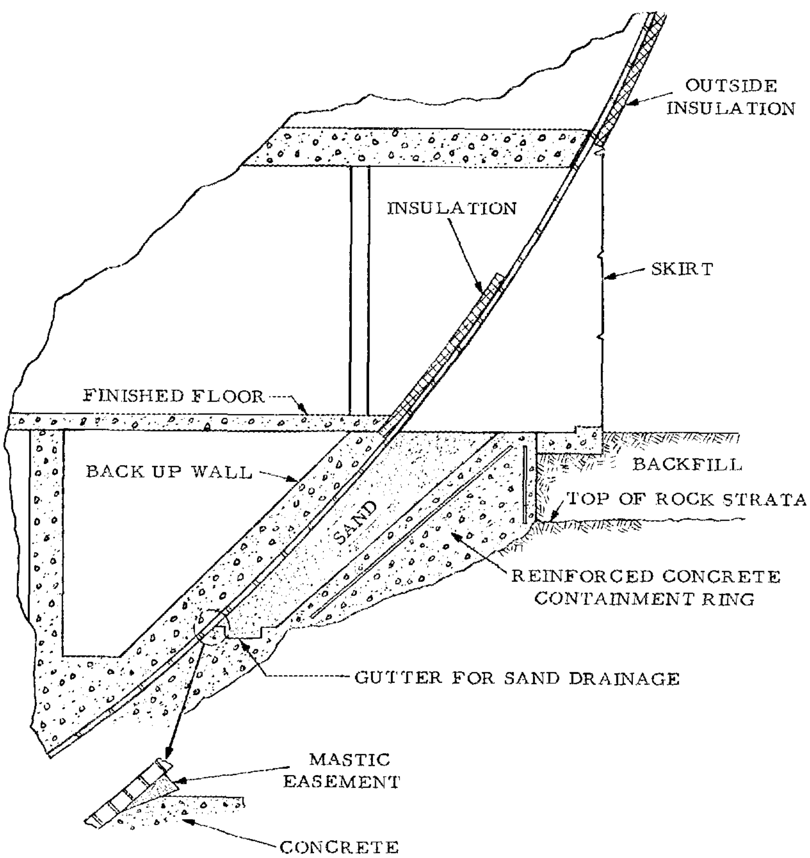



Fig. 8.18. Containment Vessel Transition Embedment at Grade. (From ref. 37, Fig. 5)

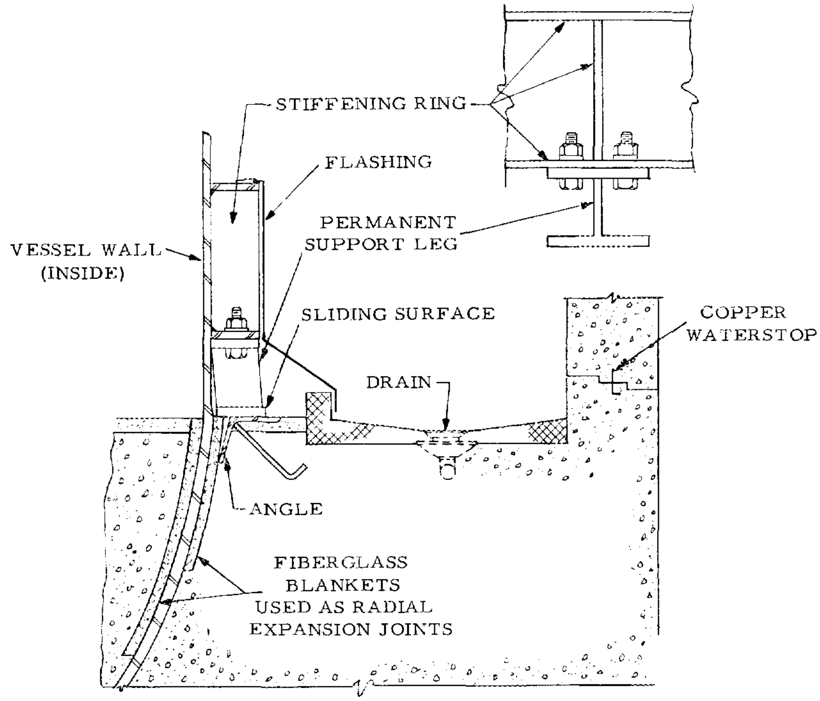



Fig. 8.19. Containment Vessel Support Leg and Radial Expansion Joint. (From ref. 36, Fig. 5.10)

### 8.3.4 Shock and Missile Protection

Shock waves could be generated by a rapid release of coolant or other fluids within a containment vessel or drywell under certain accident conditions, and therefore it may be necessary to consider the effect of the shock on the vessel. Usually the biological shielding around a reactor primary system is sufficiently massive and located so that it would tend to dissipate a shock wave originating in the reactor system and prevent it from reaching the containment shell, although some missiles might be generated. This can be seen in the cross-section drawings of several plants included in Chapter 7. Some shock-loading conditions were assumed for the Enrico Fermi containment vessel but proved to be less than the design internal pressure.[36] More detailed information on the source of shock waves and their magnitude is presented in Chapter 6.

Missile protection for steel containment vessels is achieved wherever possible by enclosing within shield walls all equipment considered capable of creating missiles. This was done at the Dresden and Big Rock Point plants. With this arrangement, care must be taken that there is an effective means of venting the shielding structure to the surrounding containment volume to avoid a dangerous local pressure buildup. Often doors and other access openings are insufficient for this purpose and additional openings must be provided. (For further information see Chapter 6.)

Concrete containment structures are considered inherently missile resistant, and special missile protection is usually not required, except as may be needed to protect metal liners. Where it is considered possible for missiles to strike the wall of a steel containment vessel, it may be necessary to line the steel with concrete. In the EBWR vessel, a 24-in.-thick concrete liner was provided below the main floor to protect the shell from missiles caused by an explosion in the nuclear system. A 12-in. concrete liner was provided above the main floor to protect against missiles from the enclosed turbine.[38] The original designs for the Shippingport containment vessel included a 6-in. gunite inner lining for missile protection.[7] The gunite was finally omitted when it was decided that it was not reasonable to assume that a failure of austenitic pipe could generate missiles.

At Humboldt Bay, the cylindrical portion of the drywell vessel membrane is solidly backed by reinforced concrete that provides strength for the drywell membrane and protection of the suppression chamber vessel against missiles that might be created by steam rushing out of a rupture in the primary system. These forces might otherwise cause penetration of the drywell membrane. The heads are lined on the inside with concrete for the same purpose.

### 8.3.5 Attachments and Accessories

Large containment structures usually require a number of attachments and accessories. Paragraph N-1342(d) of Section III of the ASME Code requires that all permanently welded attachments to the shell be impact-tested material unless they are to be heat treated after welding. Rules are also given for the welding of nonimpact-tested material to permanent attachments. To meet these requirements and also to provide acceptably low stresses, circular pad plates of A-300 steel are often welded to the shell at the attachments.

Steel eyes are frequently welded to the inside of a steel dome to support temporary maintenance scaffolds. In the Dresden containment vessel, eyes are spaced on approximately 8-ft centers in each direction and are designed for 1000 lb each. Interior platforms are not normally permanently attached to the containment vessel wall. Other examples of welded attachments are welded studs for the support of insulation and stiffening members to protect against externally applied collapsing loads.

The accessories required for inspection and maintenance of the exterior surfaces of steel containment vessels vary greatly. The Dresden vessel was originally provided with a fixed exterior stair and ladder arrangement, and it was anticipated that maintenance could be accomplished by means of a boatswain's chair tied to a nozzle at the top of the sphere. When an extensive program of insulation repair was decided upon, a rolling scaffold was installed that covered the top hemisphere from the equator to the 60th parallel. This scaffold runs on rails bolted to permanently welded, shell-attachment pads and is propelled by a hand crank. A similar movable scaffold was incorporated in the design of the containment sphere at Big Rock Point. It can be seen clearly in Figs. 8.22 and 8.24 of Section 8.4.1. In this case, the scaffold itself provides access to the top of the sphere. For the vertical cylinder containment vessel at Peach Bottom, a moving scaffold covers the top knuckle with provision for attaching a boatswain's chair to cover the vertical sides. In all these cases the scaffold terminates at a safety fence at the 60th parallel.

Opinion regarding the necessity of lightning rods on large steel containment vessels varies, but the trend appears to be away from their use. The National Fire Protection Association Code for Protection Against Lightning (NFPA No. 78-1959)[39] allows the use of structural steel framework as the main conductor for lightning protection systems providing it is grounded at distances not exceeding 60 ft apart at the perimeter. This code also acknowledges that properly grounded water tanks need not have protection if lightning will not damage the wall material. Specifications on grounding and lightning arrestor tip materials and dimensions are also provided. A lightning rod was installed on the Fermi containment vessel, although tests showed that it was virtually impossible for a lightning bolt to puncture the vessel.[36]

A cylindrical containment vessel often lends itself to supporting a bridge crane from its walls. However, the brackets supporting the crane girder impose bending moments and other loads on the shell, and for that reason the girder is sometimes structurally separated from the containment shell. For example, the 150-ton bridge crane at the Enrico Fermi plant is supported on a continuous ring attached to the containment vessel wall, whereas the circular girder for the 25-ton crane at HWCTR is supported on separate columns at the periphery of the containment vessel. In spherical vessels, cranes are invariably supported on the internal shielding structure.

Occasionally, a water tank is suspended from the roof of the containment vessel to provide a positive head source of water for emergency reactor cooling. This tank is sometimes an integral part of the roof of the containment vessel. Such is the case in the EBWR, Elk River, and HWCTR containment vessels. A figure showing this detail is included in Chapter 7.

## 8.4. CONSTRUCTION

Since nearly all steel containment structures are fabricated in accordance with the ASME Code for Unfired Pressure Vessels and related Code Cases, the details of construction are essentially those that have been used for many years. However, because of the size of these vessels, industry has had to develop new approaches to field erection to ensure economical production and still maintain adequate quality control. To illustrate the special problems involved, it is helpful to follow a typical construction sequence for both single-stage and multiple-stage erection.

### 8.4.1 Single-Stage Construction

Single-stage construction is defined as the erection of the containment vessel as a complete unit prior to construction of any enclosed facilities. After pressure and leakage rate testing, one or more large construction access ports are usually cut in the containment vessel wall and the additional foundations, plant structure, piping, and equipment are then installed. After the plant has been completed and the construction openings have been sealed, a final leakage rate test is performed to ensure compliance of the final structure with the leakage rate specifications.

A recently completed typical example of single-stage construction is the containment vessel of the Big Rock Point plant, which is illustrated in Figs. 8.20 through 8.24. The containment vessel is a 130-ft-diam steel sphere erected on 14 pipe columns, which support the sphere at the equator. The construction sequence is indicated in Figs. 8.20 through 8.24.

After excavating, the equator columns were set, and the entire vessel was erected, complete with locks and capped penetrations. The welds were fully radiographed. (See Sec. 8.4.4 for discussion of fabrication and erection techniques frequently employed.)

The completed sphere was pressurized to 5 psig, soap-bubble tested, and then pressurized to 1.25 times the design pressure and held for 1 hr. The pressure was then reduced to the design value (37 psig), and an integrated leakage rate test was conducted using the reference vessel method. The testing sequence was essentially the same as that described in Chapter 10.

After the leakage rate test, a 24- by 24-ft construction access hole was cut at ground level. Interior and exterior foundations were placed by making alternate pours. Then the internal plant structure and the

Fig. 8.20. Big Rock Point Containment Vessel Showing Temporary Support Columns.

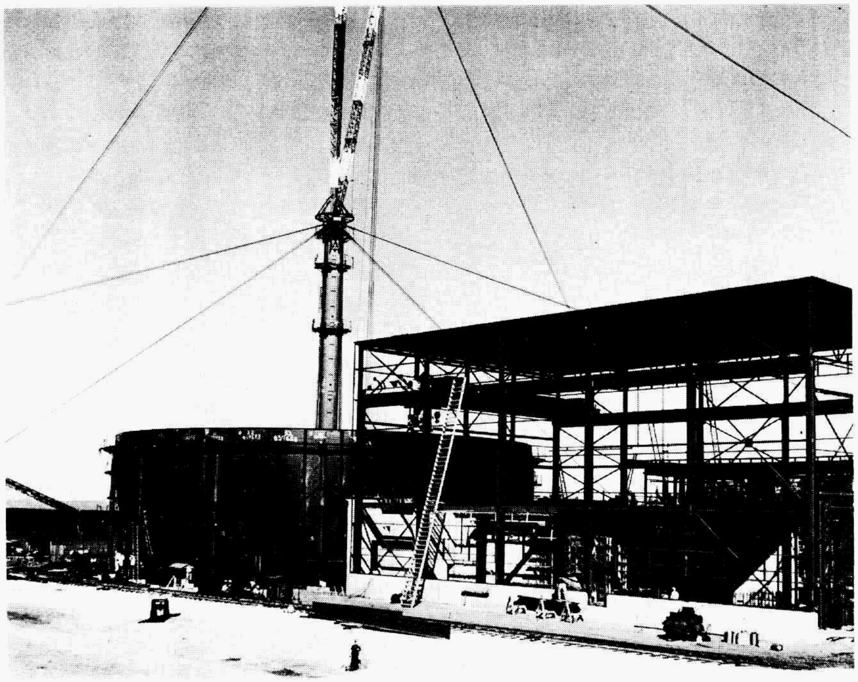


Fig. 8.21. Big Rock Point Containment Vessel During Erection.

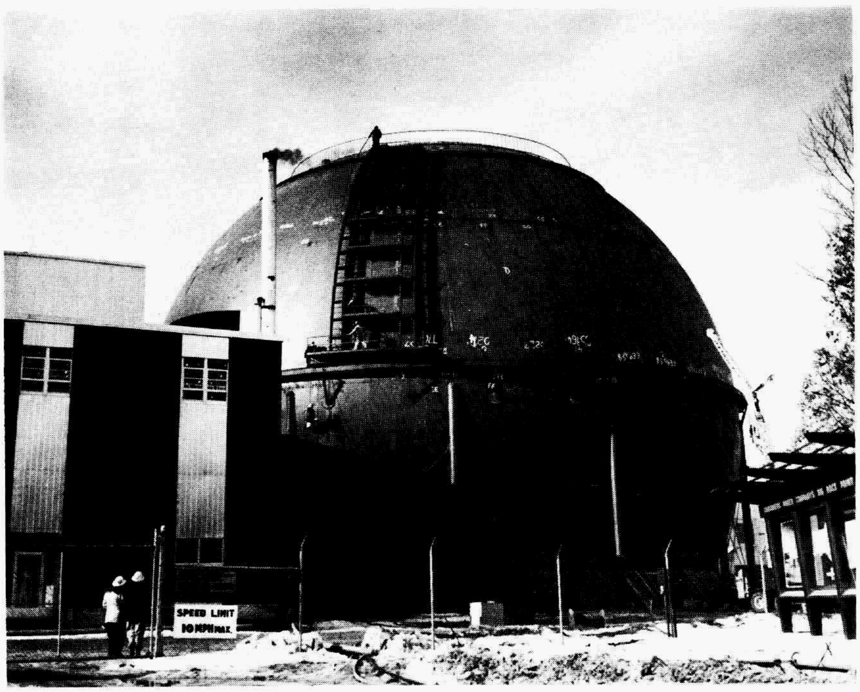



Fig. 8.22. Soap Bubble Testing of Big Rock Point Containment Vessel for Leaks.

Fig. 8.23. Big Rock Point Reactor Pressure Vessel Being Inserted into Containment Vessel Through Temporary Construction Access Hole.

Fig. 8.24. Completed Plant at Big Rock Point.

equipment were installed; the penetration caps were removed; the piping and electrical cables were installed, routed through the penetrations, and sealed; and the construction access opening was closed. The original 14 structural support columns were left in place but loosened so that they would not produce stresses by suppressing normal vessel movements.

The sphere was then pressurized to 5 psig and electrical penetrations were halide leak tested. The piping penetrations and locks were soap-bubble tested. The sphere was then leakage rate tested at a pressure of 10 psig by the same procedure as was used earlier. At 10 psig the electrical penetrations were again halide leak tested, and the locks and piping penetrations were soap-bubble tested.

### 8.4.2 Multistage Construction

Many new plants have taken advantage of a 1956 ruling of the ASME Boiler and Pressure Vessel Committee (Case 1228), which permits partial erection of the containment vessel prior to construction of the supporting foundation, internal structure, and equipment. Case 1228, which has since been included in Case 1272N, states that

> "Inspection of welded joints in the lower part of containment vessels during the pneumatic test will be waived where such joints are covered by concrete during the construction of the vessel, provided:
>
> "a) There are no openings or penetrations of the part of the vessel covered by concrete, and
>
> "b) All welded joints that are inaccessible for inspection during the test of the completed vessel shall be Type No. (1) of Table UW-12 [double butt type] and shall be fully radiographed and prior to being covered shall be tested for leak tightness using a gas medium such as Halide Leak Detector Test."

The same provision is included in paragraph N-1411 of Section III of the ASME Code. This ruling may have a beneficial effect on the overall job cost, since it permits the construction of the interior portions of the plant by conventional methods of material and equipment handling and may permit more efficient utilization of construction forces.

The construction of the 100-ft-diam, 162-ft-high capsule containment vessel used for the Peach Bottom Atomic Power Station is a typical example of multistage construction. It is illustrated in Figs. 8.25 through 8.29.

After excavating, the lower ellipsoidal head was erected to approximately 3 ft above grade, welded, radiographed, and halide leak tested. The interior and exterior foundations were then placed; the plant structure was completed; and the bulk of equipment and piping was installed. A 200-ton capacity guyed derrick was placed atop the concrete structure for erection of the reactor vessel, helium dump tanks, steam generators, and helium compressors.

The cylindrical portion and upper ellipsoidal head of the containment vessel were then erected, welded, radiographed, and soap-bubble tested.

Fig. 8.25. Initial Construction of Lower Portion of Containment Vessel at Peach Bottom.

Fig. 8.26. Testing of Lower Ellipsoidal Head of Peach Bottom Containment Vessel.

Fig. 8.27. Placement of Concrete Under Lower Head of Peach Bottom Containment Vessel.

Fig. 8.28. Simultaneous Placement of Concrete Inside Lower Head of Peach Bottom Containment Vessel.

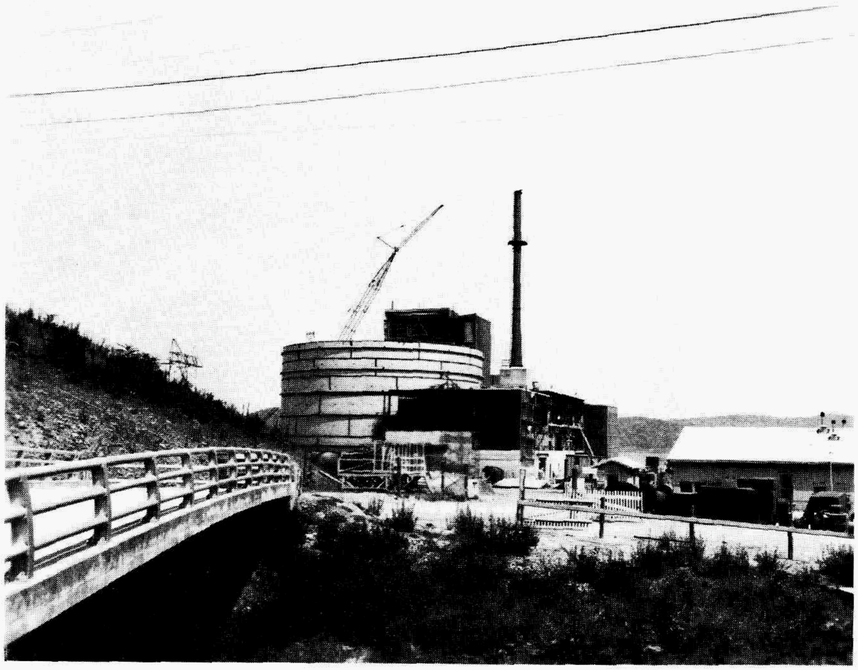


Fig. 8.29. Erection of Containment Cylinder Walls and Internal Equipment at Peach Bottom.

A pressure test at 1.15 times design pressure was conducted followed by an integrated leakage rate test at design pressure.

Penetration caps were then removed, and all cables and piping were routed through the penetrations and sealed. All work done subsequent to the previous leakage rate test was soap-bubble tested or halide leak tested. The vessel was then given its final leakage rate test. Either the single-stage or the multistage construction method can be used for most containment designs and the method to be used has little effect on the choice or shape of vessel. However, in composite containment structures, such as the HWCTR vessel,[25] the construction is multistage by the nature of the different materials used. The comparative advantages and disadvantages of single- and multistage construction are outlined in Table 8.12.

### 8.4.3 Lined Concrete

In concrete containment structures, the concrete must usually be lined with metal or a nonmetallic coating to provide the necessary degree of leak tightness. If a high degree of leak tightness is required, a welded and fully tested metal liner is essential. A coating also may be required to provide an easily decontaminated surface.

Table 8.12. Comparative Advantages and Disadvantages of Single- and Multistage Construction Sequences

| | Advantages | Disadvantages |
|---|---|---|
| Single stage | 1. Lower construction cost for the vessel | 1. Requires temporary interior lighting and ventilation |
| | 2. 100% observation of welds under pressure | 2. High noise level during interior construction |
| | 3. Provides weather protection for interior construction | 3. Limited access |
| | | 4. Difficult rigging of materials and equipment |
| | | 5. Congestion raises construction cost of interior work |
| Multistage | 1. Free access for erecting interior plant | 1. Higher construction cost for the vessel |
| | 2. Overall cost of combined structure and vessel is normally less than that for single-stage construction | 2. Separate leak testing required for bottom portion of the vessel |
| | 3. Temporary interior lighting or ventilation may not be required | 3. Piping and electrical connections must be delayed due to later completion of containment vessel |

#### 8.4.3.1 Metal Linings

The metal linings used are of two types: (1) those attached by welding to cast-in-place inserts in the concrete, such as used at Hallam and illustrated by Fig. 8.30, and (2) those used as integral forms for the concrete, such as used at Humboldt Bay and illustrated by Fig. 8.31.

Since the lining is used only as a leaktight membrane and is not a pressure vessel, its fabrication does not fall under Section VIII of the ASME Code. To maintain the integrity requirements, however, it is still necessary to require welding quality control and leak testing.

Extensive use of liner plates welded to inserts was made on the Hallam Nuclear Power Facility containment structure. This method was used because it was desirable to be able to vent any moisture trapped

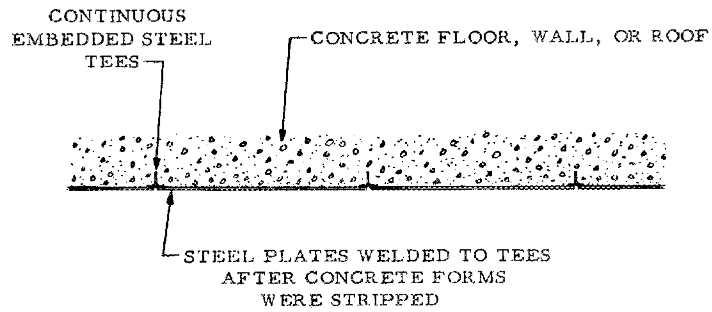



Fig. 8.30. Linear Plate Detail for Subterranean Cells at Hallam Nuclear Power Facility.

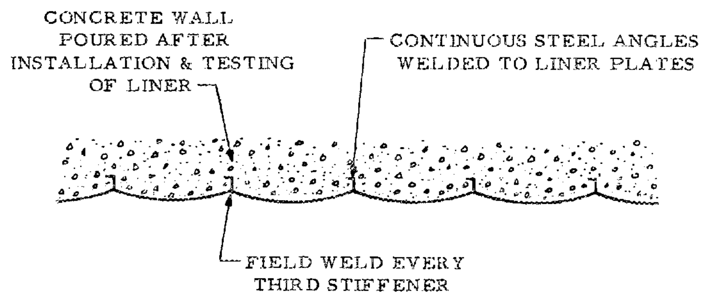



Fig. 8.31. Linear Plate Detail for Suppression Pool Walls at Humboldt Bay Nuclear Power Facility.

in the space behind the liner since the cells were to operate at temperatures exceeding 212°F. Structural tee inserts were embedded on centers that varied from 2 ft 9 in. to 6 ft 0 in. The 1/4-in. floor plate and 12-ga. wall and ceiling sheets were seal welded to the inserts immediately following the removal of the concrete forms. All welds were soap-bubble tested at 3 psig with vacuum boxes. After installation of the piping, cables, and equipment, the structure was given an integrated leakage rate test.

The suppression pool of the Humboldt Bay Power Plant (see Figs. 8.1 through 8.5) is an example of the use of an integral liner. The exterior wall of the suppression pool was constructed as a caisson, being sunk to a depth of 78 ft below grade after being constructed in four lifts at grade. Since waterproofing materials could not be expected to stay in place as the caisson was sunk, the liner was designed to resist the pressure of the external ground water, which will eventually leak through the 4-ft-thick concrete caisson wall. The internal pressure and the external soil pressure are carried by the concrete structure, with the liner acting as a leaktight membrane. The liner-plate segments were fabricated with vertical stiffeners on 4-ft centers, with the plate scalloped inward between stiffeners to transfer the external ground-water pressure load to the stiffeners. Concentrating the form ties at the vertical stiffeners allowed the prefabricated segments of liner plate to be used as the concrete forms. The liner plate segments were welded, 10% spot radiographed, and 100% soap-bubble tested prior to erection of other forms. The concrete was placed, the wood forms stripped, the caisson sunk level with the ground, and the next 15-ft lift formed and placed. After sinking of the caisson was completed, the bottom was sealed with concrete and pumped dry. A level floor of concrete was then placed in the bottom of the suppression tank and lined with steel plate welded to embedded steel. These plates were soap-bubble tested only. The roof of the suppression tank was formed of a 3-ft-thick concrete slab lined with steel plate welded to the wall liner plate. Following leak testing, the roof liner plate was also used as a form after being stiffened and supported by steel beams. Upon completion, the suppression pool was pressure tested at 20.3 psig and leakage rate tested at 10 psig using the reference-vessel method.

It can be assumed that metal linings will find increasing use because of widespread interest in the following types of containment:

1. Pressure suppression,
2. Pressure relief,
3. Composite type (concrete base and steel hemispheres),
4. Low-pressure containment,
5. High-pressure concrete vessels.

In the containment system proposed for the Malibu plant at Los Angeles[40] and for other large nuclear plants to be located near metropolitan areas, two welded steel membranes are used. The thin steel plate membranes are separated by a reduced pressure zone filled with porous concrete, and the entire sandwich is backed up by an external concrete pressure-containing structure that also serves as a containment shield. The double membrane allows continuous monitoring of the intermediate zone for leaks and provides a mechanism for controlled venting of all leakage from inside the containment vessel or the return of the leakage to the containment vessel.

The principal advantage of a metal liner as a leakage barrier is that it allows the concrete structure to carry the pressure at normal working stresses without concern about leakage through stress cracks or existing shrinkage cracks. Supplementary advantages are that it can be used as a concrete form and may also be utilized as a waterproofing membrane in some underground locations. As with any membrane it is necessary to consider the possibility of deformation or buckling due to external pressure or adverse temperature conditions. Consideration should be given to corrosion protection and specimen monitoring in those instances where membranes are exposed to water seepage and alternating evaporation and condensation conditions.

### 8.4.3.2 Nonmetallic Coatings

In general, coatings are used as leakage barriers for concrete structures, where some leakage is tolerable, with the realization that some maintenance may be necessary during the life of the plant because of shrinkage or settlement cracks and damage due to abrasion or impact of coating surfaces.

Coatings that are intended to serve both as leakage barriers and decontamination coatings are usually of the 100% solids type (phenolics, polyesters, epoxys, etc.). These types of coatings can withstand traffic and abrasion but are not as flexible or self-healing as asphaltic or coal-tar pitch coatings.

Where coatings are to be left exposed, they are usually applied after major equipment and piping have been erected to prevent serious construction damage. It is also necessary to protect floor coatings during construction by decking with an inexpensive plywood or heavy cardboard.

A rigid coating of "Liquid Tile" (a thermal-setting nonporous plastic resin) was used on the concrete portion of the HWCTR containment vessel.[25] Waterproofing of the exterior surface below grade to obtain a dry wall and special etching and cleaning of the internal concrete surface were necessary to ensure a proper bond. The "Liquid Tile" was applied to all interior surfaces of the concrete containment wall and floor, to a 3-ft return on all walls and floor slabs abutting the containment vessel, and to the top and bottom surfaces of the ground-floor slab.

The effective use of a flexible, builtup bituminous and pitch membrane as a vapor barrier was demonstrated on the BONUS reactor building.[41] The 182-ft-diam concrete-base slab is 3.5 ft thick and is surfaced with a bituminous paint and a laminated membrane consisting of three 3/16-in. plys of coal-tar pitch and two plys of asbestos felt. The membrane is covered with 2 ft of sand topped by a 6-in. concrete floor slab. All construction joints in the lower slab have waterstops and are sealed at the top by "Jet Careylastic" joint compound and coal-tar pitch. In addition, three coats of "Bituplastic 33" (a coal-tar polymer emulsion) were extended up the walls to form a longer leakage path. These types of materials have the advantage of low cost, ease of application, low permeability, ability to self heal, and resistance to deterioration. Disadvantages are poor wear resistance and difficulty of decontamination.

### 8.4.4 Fabrication and Erection Techniques

The large amount of field welding involved in large steel containment vessels has necessitated the development of production techniques for using large fit-up jigs and the automatic welding equipment normally used only in the shop. On the Dresden sphere,[37] for example, there are over 17,000 lin ft of double-butt-welded joints in plates varying in thickness from 1.25 to 1.40 in. The shell plates were shipped precut and dished in sizes up to 10 by 35 ft. To minimize hand welding, these large plates were machine welded on large custom-built tables. The tables were adjustable and could accommodate four plates at once. Nearly 60% of the field welding was done by the automatic submerged-arc process. The welding tables saved considerable time and expense and also gave good quality control. The machine welds were 100% radiographed and repaired on adjacent tables prior to erection.

Erection of the plates required a 50-ton capacity derrick for unloading and subassembling the plates, a 100-ton guyed derrick mounted centrally in the sphere to erect the subassembled plates, and various booms and hangers supported on the interior derrick tower for maintaining spherical alignment of the erected plates during fit-up and welding. All welding in the erected position was done by hand with low-hydrogen electrodes.

All joints were double butt welded, the back weld being made after carbon-arc gouging and magnetic particle inspection of the first weld. Full radiography of the welds was done as the welding progressed.

The shop fabrication for the Dresden sphere consisted of (1) shaping, trimming, beveling edges, pickling, and painting of shell plates; (2) welding penetration nozzles and column stubs to shell plates and heat treating the assembly; and (3) fabrication and postweld heat treatment of locks. One large bolted door, with a 16-ft-diam clear opening, that was mounted in a 26-ft-OD insert plate required field welding and postweld heat treatment. To minimize distortion between the gasketed surfaces of the door, the welding was done on a round-the-clock basis and followed a carefully detailed sequence with magnafluxing of many interpass beads. The door was bolted rigidly to the insert plate during the welding and the postweld heat treatment (1150°F for 3 hr).

The fabrication and erection techniques for steel containment vessels of other shapes are similar to those mentioned above. In general, every effort is made to handle the unloading, subassembling, erection, welding, and testing of the pieces in a production-line manner, to standardize procedures, to reduce field time and cost, and to increase quality control of welding and tolerances. Field erection of most containment vessels requires specially fabricated derricks, welding tables, handling devices, and erection jigs, which comprise a sizeable portion of the total cost of the job.

Little effort has been made to specify erection tolerances other than those contained in the Unfired Pressure Vessel Code.[3] However, the code tolerances were established for much smaller vessels and are not applicable to most containment vessels. The Unfired Pressure Vessel Code (Par. UG-80a) permits an erection tolerance of up to 1.0% on the diameter of cylindrical shells, allowing a 100-ft-diam cylinder to be out of round as much as 12 in. Section III of the ASME Code[2] contains no additional provisions for tolerances that are applicable to containment vessels.

The principal problem created by loose tolerances, that of avoiding contact with adjacent structures, has been alleviated in most cases by the rigid control and tighter tolerances actually achieved by the fabricators in erecting the vessels and by allowing relatively large clearances between the vessel and the interior structures, except at grade. However, in recent multistage capsule-type vessels, such as the vessel at Peach Bottom, the building structure and platforms extend to within 6 in. of the vessel wall and are erected prior to vessel erection. Because existing standards were inadequate or unavailable, it was necessary for the designer to include the tolerance requirements in the job specifications.

## 8.5 BRITTLE FRACTURE OF STEEL

J. R. Hawthorne H. B. Piper
L. E. Steele

The brittle failure of large steel structures[42] is a widespread problem that has attracted the attention of engineers and scientists.

The dramatic failures of merchant ships that followed the advent of welded ship construction in the early years of World War II have been widely discussed,[43-45] as have many brittle failures of bridges, pressure vessels, pipelines, and storage tanks.[46] The investigations and analyses that followed as the aftermath of these incidents showed that the origin of the failures lay in the tendency of a broad class of metals and alloys to exhibit low-temperature brittleness. This tendency is increased as a result of exposure to radiation, and hence the possible embrittlement of reactor pressure vessels in service is a matter of concern from the standpoint of nuclear safety. Furthermore, the containment vessel may be subject to failure from either the influence of ambient temperatures or neutron exposure. Both these possibilities are of interest to the containment designer. The information presented here is directed toward failure of the reactor pressure vessel, since this vessel will receive a large neutron dose during its lifetime, but the principles involved may be extended to cover other pressure-containing systems as well, such as the containment shell.

The purpose of the following discussion is to familiarize the reader with brittleness of steels, the tests used to determine this brittleness, the effect of neutron exposure on brittleness, and surveillance and remedial techniques that might be used to minimize the hazards of a brittle rupture.

## 8.5.1 Transition Temperature Tests

The energy that is absorbed before fracture takes place in the low-strain-rate tensile test may be used as a measure of ductility or, more precisely, "toughness." For steels the ductile-to-brittle transition for this type of test would occur at quite low temperatures. On the other hand, when impact loading of notched samples is employed, the transition temperature is higher. Notch-impact tests have therefore been devised in which the resistance to fracture is measured in a more convenient temperature range. Also, the transition temperatures that are measured in notch-impact tests are more representative of temperatures at which service failures have taken place. In the testing of reactor-pressure-vessel steels, the Charpy V-notch impact test is probably in widest current use, but other types of tests are also employed, such as the drop-weight test,[47-50] the explosion bulge test,[47,48,51] and the crack-arrest test.[47,52,53] These four tests serve to determine the nil-ductility transition (NDT) temperature, the fracture transition elastic (FTE) temperature, and the crack-arrest temperature (CAT). These quantities play a part in the analysis of service failures and the establishment of design criteria.

### 8.5.1.1 Impact Tests

In the Charpy and Izod impact tests[47,54] a notched sample bar is struck a single blow by a swinging pendulum, and the fracture energy is measured by the reduction in the energy of the pendulum upon breaking the sample. The test is repeated on a number of identical samples at various temperatures, and the fracture energy is recorded as a function of test temperature. A typical impact curve is shown in Fig. 8.32 for Charpy V-notch samples of a pressure-vessel steel.[55] For this material and type of test, the fracture energy is high (40 ft-lb) for temperatures above 0°F and low (less than 10 ft-lb) below −100°F. Since the curve of fracture energy versus temperature is not a perfect step function, a fracture energy value must be chosen for which the ductile-to-brittle transition temperature is specified.

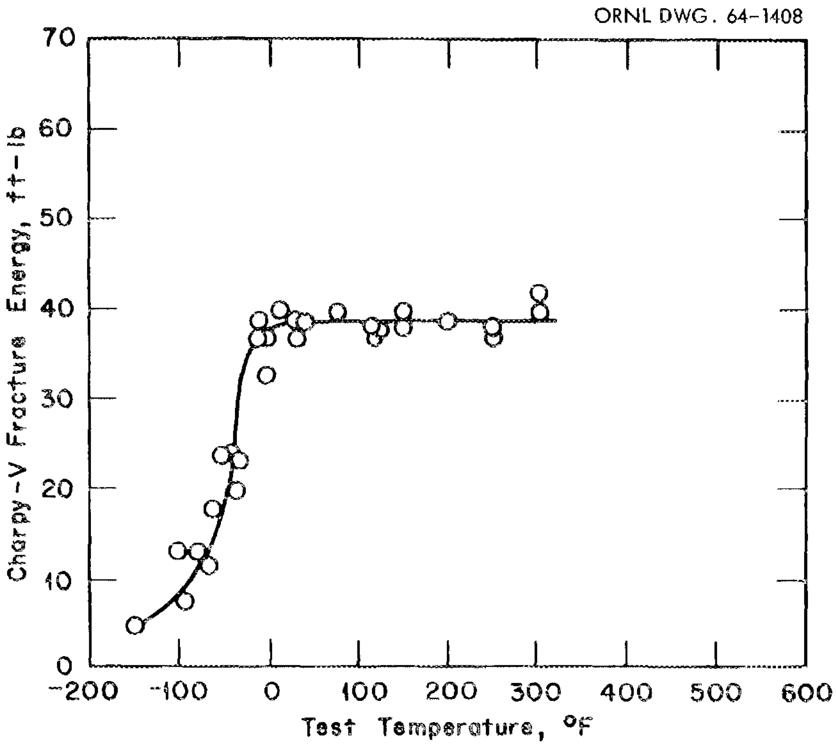



Fig. 8.32. Charpy V-Notch Impact Energy Curve for ASTM A-212 Grade B Steel Normalized to 1700°F. (From ref. 47)

8.5.1.2 Drop Weight Test

In the drop-weight test, the specimen is struck by a falling weight, and a crack is started in a small brittle weld bead on one of the plate surfaces. Stops are arranged so that the surface on which the bead is welded is stressed in tension to the yield stress. The NDT temperature is determined by the temperature at which the sample breaks completely. Thus the drop-weight test is a "go, no-go" test that determines the temperature at which a small flaw will propagate through the specimen and produce a brittle fracture. As a result of tests on ship-fracture steels, it was concluded[56] that the NDT temperature represents the highest temperature for the initiation of brittle fracture in conventional steel structures.

Despite the fact that the term "NDT temperature" is defined specifically with reference to the drop-weight test, it has become common practice to use the term more generally to indicate the ductile-to-brittle transition temperature, as determined by other tests. For the Charpy V-notch impact test, it is possible to correlate the drop-weight NDT with the temperature corresponding to a certain Charpy V-notch energy. This energy varies somewhat from one steel to another. For reactor pressure vessel steels, such as A-201B, A-212B, A-320B, A-350-LF-1, A-350-LF-3, A-336, and A-105, this correlation energy is considered[48] to be in the range of 20 to 30 ft-lb. Because of the smaller size of the Charpy V-notch sample (75 × 10 × 10 mm) as compared with the drop-weight sample (14 × 3.5 × 3/4 in. or 5 × 2 × 5/8 in.), the Charpy sample is considerably more convenient for reactor experiments. Therefore, in many cases in which the results of irradiation are quoted as "changes in NDT temperature," the changes have generally been measured by use of notched-bar impact tests and not drop-weight tests.* Irradiation experiments[48] have been carried out using drop-weight samples, however, in which it was demonstrated that the shift in the Charpy V-notch transition temperature (30 ft-lb level) caused by irradiation was equal to the shift in the drop-weight NDT temperature.

### 8.5.1.3 Explosion Bulge Test

The test plate for the explosion-bulge test[47,48,51] on which a brittle crack-starter weld is applied is 14 in. square by about 1 in. thick. The plate is explosively loaded so that the weld bead is placed in tension. A hold-down die is used so that the edges of the plate are only elastically loaded. The FTE temperature is taken to be the highest temperature at which the fracture propagates through this elastically loaded die-supported region. The temperature that corresponds to the NDT temperature is reached at approximately 60°F below the FTE temperature for pressure-vessel steels. Below this temperature the plate breaks without bulging (i.e., without visible plastic deformation). Thus the FTE temperature is the temperature above which a crack will not propagate through a region of the sample that is stressed at or below the yield stress.

### 8.5.1.4 Crack-Arrest Test

The crack-arrest, or Robertson, test[52,53] involves the use of a large rectangular plate that is at least 12 in. wide. The specimen is subjected to a uniform tensile stress. In one variant of the test, the test piece is held in a temperature gradient, and a rapidly running crack is started across the test piece from the cold end. At a given stress level, the temperature is observed that corresponds to the point at which the shear lip thickens or the crack stops. This CAT as a function of the applied tension stress determines the CAT curve. The CAT for loading at the yield

*In this discussion, changes in notched-bar ductile-to-brittle transition temperatures are referred to as changes in NDT temperature.

stress is correlated with the FTE temperature. In a subsequent discussion of the actual reactor pressure vessel, it will be seen that the fact that the FTE temperature is approximately 60°F higher than the NDT temperature for reactor pressure vessel steels[57] plays a part in the design criteria for reactor pressure vessels. In another modification of the Robertson test, the test piece is held at a uniform temperature and a "go, no-go" test is applied in which the crack either succeeds or fails to propagate entirely across the plate at a given applied stress.

### 8.5.2 Low-Temperature Brittleness

A characteristic feature of the metals that exhibit low-temperature brittleness is the presence of a ductile-to-brittle transition over a rather narrow range of temperatures. The temperature at which the ductile-to-brittle transition takes place is not an inherent constant of the material. Instead, the ductile-to-brittle transition temperature depends upon the type of test used to measure it and specimen geometry. The transition temperature usually increases with the loading rate of the test and the triaxiality of the stress state to which the sample is subjected. Many tests have been devised to evaluate the ductile-to-brittle characteristics of metals. They may be classified[42] according to the type of loading (tension, bending, tension plus bending) and according to whether the samples are welded or notched.

Above the transition temperature, upon the application of higher and higher tensile stresses, the metal undergoes elastic distortion until the yield stress is reached. It then exhibits uniform plastic deformation until a mechanical instability takes place in which the metal suffers a localized reduction in area, or "necking." Finally the metal fractures at the neck as a result of a tearing or shearing action. This ductile behavior entails the absorption of a considerable amount of energy prior to fracture. In contrast, when the metal is stressed below the transition temperature, very little plastic deformation takes place. The fracture is largely the result of cleavage, for which relatively little energy is required. In brittle fracture the fracture occurs as a result of the initiation and propagation of a clevage crack. With respect to the integrity of large structures, the insidious nature of the brittle-fracture process lies in the fact that, once the crack reaches a large enough size and acquires a sufficiently high propagation speed in a susceptible portion of the structure, it can continue at high speed into other parts of the structure and may result in sudden, catastrophic failure.

#### 8.5.2.1 Three Conditions for Fracture

The many investigations of brittle fracture in supposedly ductile structures consistently point to three conditions that must be present to cause such failures: (1) temperature near or below the NDT, (2) nominal stress level >5000 to 8000 psi, and (3) a stress concentration or flaw that raises the local nominal stress above the yield point.[58] Fracture-safe operation depends on properly controlling or eliminating one or more of these factors.

There is no single value that is the brittle-to-ductile transition temperature for carbon steels. The NDT temperatures established by tests can only be considered as reference points for comparative purposes, since the mode of failure of a structure, either brittle or ductile, will depend to a certain degree on its state of stress. Therefore the stresses in pressure vessels must be considered in conjunction with the NDT characteristics; areas such as the nozzle regions must also be carefully examined because of stress-concentrating factors that may prevail in those locations.

Fracture safety usually is achieved by controlling the temperature, that is, by selecting a steel with an NDT below the lowest expected operating temperature and by limiting stresses when operating temperatures are below some reference level related to the NDT — customarily NDT + 60°F. The addition of 60°F to the NDT temperature is not simply a safety factor to account for engineering uncertainty. It represents a substantially greater fracture toughness, termed the fracture transition for elastic loading (FTE), which is the temperature at which running cracks are arrested in laboratory specimens that are loaded to the yield stress. A recently proposed fracture theory provides a rational basis for using the FTE or NDT + 60°F as the temperature criterion for limiting stress.[59] This theory, which relates fracture stress, flaw size, and temperature, correlates well with the fracture histories of several structures that failed in service. The fracture diagram (Fig. 8.33) is the key to the

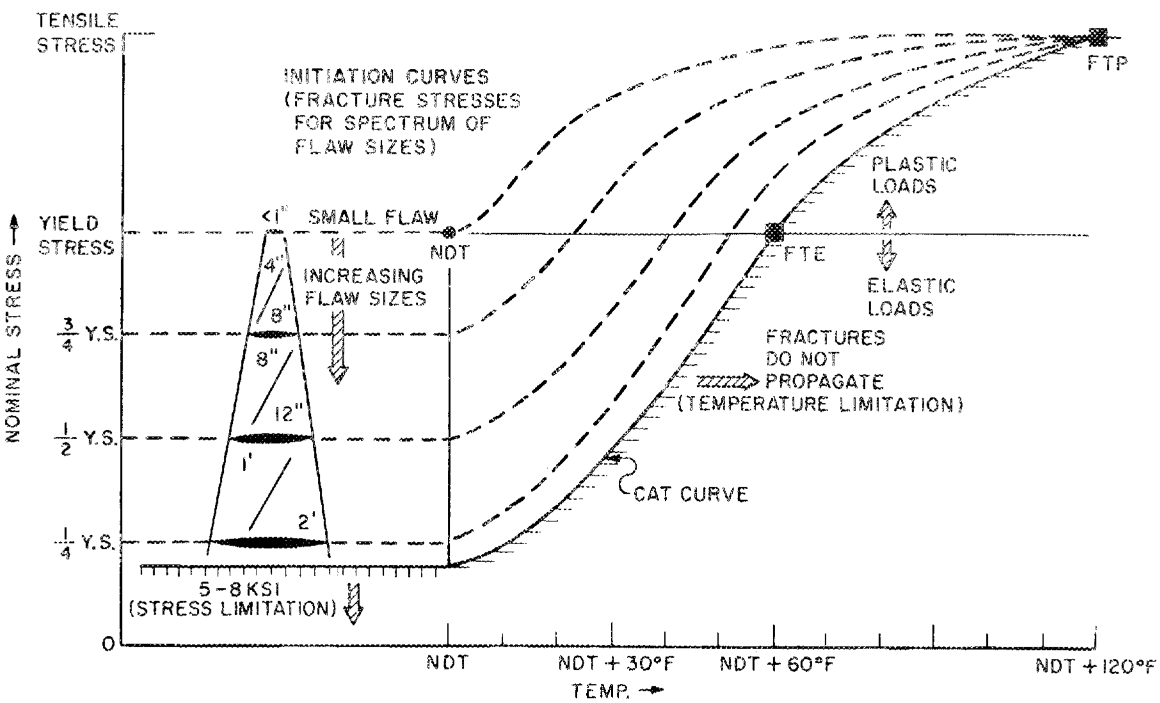



Fig. 8.33. Generalized Fracture Analysis Diagram as Referenced by the NDT Temperature. (From ref. 61)

analysis. To trigger a fracture in a flaw-free structure at the NDT temperature or below requires that stresses be at (or just above) the yield point. For successively larger flaws, at NDT the fracture stress decreases roughly in inverse proportion to the square root of the flaw length, until a lower plateau of 5000 to 8000 psi is reached; below that there is not enough elastic energy stored in the structure to support the propagation of even a very large crack. Above the NDT, the same trend of decreasing strength with increasing flaw size is evident but at much higher stress levels. At a point approximately 60°F above the NDT the stress at which even the largest crack cannot propagate rises to the yield point of the steel, and small-flaw or no-flaw strength approaches the ultimate tensile strength. The "extra" 60°F then provides substantially greater fracture resistance than the NDT criterion, thus ensuring against the propagation of an extremely large flaw at normal operating stresses. Reduction of the nominal (working) stress when below the NDT + 60°F temperature gives the fracture safety needed for operation at lower temperatures.

One approach to the problem of determining allowable stress levels with respect to NDT temperatures is presented in the tentative "Navy" code, as follows:[60]

> "In order to decrease the probability of brittle fracture of ferritic steels, the following load restrictions shall be used in relation to the NDT temperature. When the metal temperature is less than 60°F above the NDT temperature, the maximum applied load, including material pressure, shall be restricted to 20% of the design value. When the metal temperature is more than 60°F above the NDT temperature, there is no restriction upon the load from the aspect of brittle fracture prevention."

### 8.5.2.2 Significance and Practical Use of the Fracture Analysis Diagram[61]

The origin of the "classical" brittle fracture problem is the decrease in fracture toughness resulting from a change in fracture mode from high-energy-absorption ductile tearing to low-energy-absorption cleavage fracture in a rather narrow range of temperatures. The full span of the transition is in the order of 120°F. However, the transition span from an "intermediate" to a "very low" level of fracture toughness, which is the transition range of engineering interest for conventionally loaded structures, occurs over a range of 10 to 60°F, depending on the stress level.

Fracture-safe design of steel structures may be based on preventing fracture initiation or on preventing fracture propagation. Until recently there was considerable contention as to the relative merits of the two approaches. The "fracture analysis diagram," Fig. 8.33, serves to unify these approaches into a coherent analytical scheme,[62] as illustrated by the generalized stress-temperature curves for both crack-arrest and fracture initiation. For temperatures above those of the crack-arrest-temperature (CAT) curve, brittle fractures are indicated to be prevented by the "crack-arrest" properties of the steel. This signifies a degree of fracture toughness sufficient to arrest the propagation of a brittle fracture translated through a brittle plate welded to a "test" steel. The CAT approach avoids questions of flaw-size and stress requirements for fracture initiation by restricting the use of steels to temperatures or to stress levels of "no propagation." This simplicity was particularly appealing prior to the development of adequate relationships of flaw size, stress, and temperature represented by the family of fracture initiation curves of the fracture analysis diagram.

The use of the diagram depends on the accurate determination of the NDT temperature. This may be accomplished directly by the standard drop-weight test[63,64] or indirectly by the $C_V$ test. The usefulness and limitations of the $C_V$ test for such correlation require clarification. There is a considerable danger that misapplications will result for steels that have not been subjected to correlation studies.

The fracture analysis diagram provides a frame of reference for engineering analysis and for the application of intelligent and engineering judgment. The family of fracture initiation curves directs attention to the "quality" aspects of the vessel with respect to possible flaw sizes and also to the effective levels of stress acting on the flaws. The increase in the stress level of the fracture initiation curves with increasing temperature above the NDT directs attention to the transition temperature of the steel. The following factors must be considered in the analysis and judgment process:

1. What is the largest flaw size expected in the vessel as fabricated? The answer to this question involves judgment based on the quality of inspection and of the control procedures used during fabrication.
2. What is the level of stress acting on the flaws considered to be present at various locations? For flaws located in the smooth cylindrical areas of the shell, the stress level is indicated by a suitable vector of the PD/2t design stress. For flaws located at nozzle openings, the effective stress level will vary with design quality. In the absence of stress relief, small flaws located in the weld regions should be considered to be subjected to near yield-level residual stresses.
3. What contributions to flaw size enlargement are expected due to cyclic (fatigue) loading? The answer to this question requires consideration of prior deductions of items (1) and (2) for positions of potential fatigue action. A small flaw that is located or developed in the region of plastic stress loading should be expected to result in low-cycle fatigue growth to a size that is established by the thickness of the section, if fracture does not terminate the growth process.

As examples of fracture propagation at temperatures above the CAT, two seamless tube air flasks were subjected to burst tests. Details of the vessels and the sharply notched, machined slits are illustrated in Fig. 8.34. Both vessels were constructed of quenched-and-treated chromium-molybdenum steel of the ASTM A336 F22 type, for which the tension test data are given in Table 8.13. The burst tests were conducted at 48°F for flask E71 and 55°F for flask E75 and resulted in totally different fracture modes, as illustrated in Fig. 8.35. The first step in the fracture process for both vessels involved the development of an elliptical bulge over the length of the flaw and the tensile "neck" rupturing of the bottom

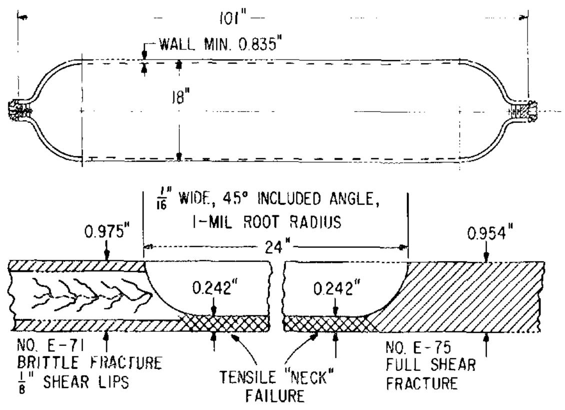



Fig. 8.34. Features of Deliberately Flawed Pressure Vessels Used for Pneumatic Load Burst Tests. (From ref. 61)

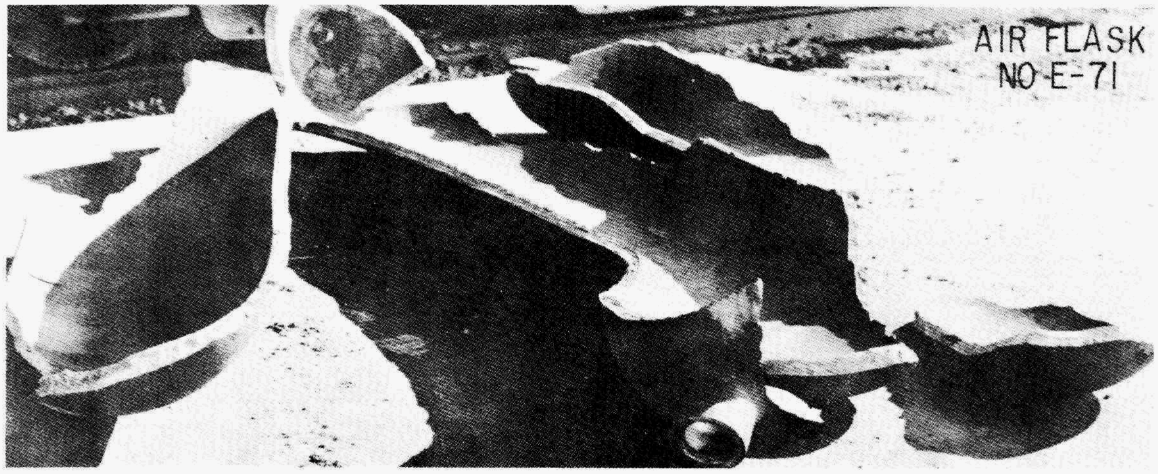
Table 8.13. Tension Test Data for 0.505-in.-diam Seamless ASTM Type A336 F22 Steel Tubing[a]

| Flask No. | 0.2% Yield Strength (psi) | Tensile Strength (psi) | Elongation in 2 in. (%) | Reduction in Area (%) |
|---|---|---|---|---|
| E71 | 84,300 | 100,900 | 75.8 | 22.5 |
| E75 | 96,300 | 111,800 | 74.4 | 20.8 |

[a]From ref. 61.

of the slit. At this point, the bulge regions "opened" and the flaw ends were subjected to very high plastic load stresses. The rupture of flask E75 started and propagated in its entirety as a 45° shear tear. That of flask E71 started and propagated as a brittle fracture with 1/8-in. shear lips, as illustrated schematically in Fig. 8.34.

The NDT of the E75 flask was −70°F, which is 125°F below the burst-test temperature. Accordingly, the fracture analysis diagram predicted that flask E75 would fracture in shear mode with an effective stress level in the order of the ultimate tensile strength of the steel. The NDT of the E71 flask was 0°F, which is approximately 50°F below its burst-test temperature. Accordingly, this flask was expected to fracture in a brittle manner with heavy shear lips and with an effective burst stress of a relatively high plastic level but below the ultimate tensile strength of the steel. The fracture modes of both vessels were predicted exactly by the NDT location of the fracture analysis diagrams with respect to the burst-test temperatures. The effective burst stresses were indicated indirectly by the degree of bulging in the flaw areas — the E75 vessel developed a considerably larger bulge than that developed by the E71 vessel.



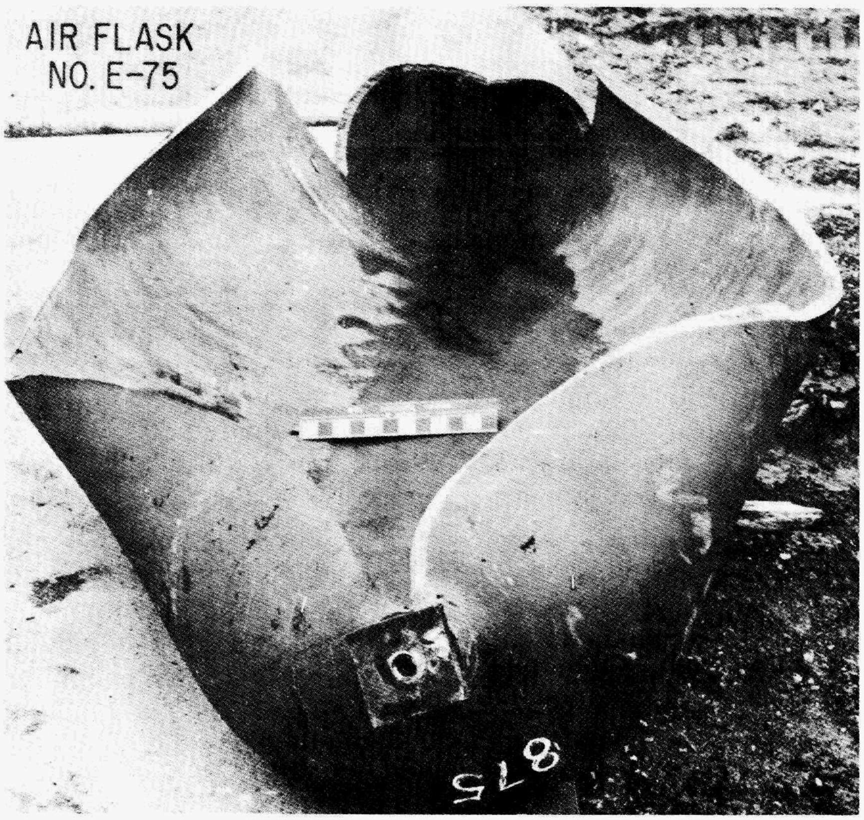




Fig. 8.35. General Mode of Failure of Pneumatically Loaded Pressure Vessels Tested in the Presence of Large Flaws. Top: brittle fracture with 1/8-in. shear lips developed by flask E71. Bottom: full 45-degree shear rupture developed by flask E75. (From ref. 61)

## 8.5.2.3 Comments Pertinent to Pressure Vessel Fabrication[61]

The rapidly developing background of information on procedures for the prevention of brittle fractures has resulted in an increasing demand for fracture toughness data for the common grades of structural steels. In effect, the design engineer has awakened to the realization that large tonnage production procedures are sufficiently reproducible so that a particular grade of steel of a given thickness range may be categorized by an expected range of NDT or crack-arrest temperatures. However, there is no published compendium of such data. In effect, the information on the use of these procedures to provide fracture-safe engineering design of steel structures has preceded the development of the necessary data.

The primary specification requirements for steels involve controls on chemical composition, tensile test properties, plate thickness, deoxidation practice, and the use of normalizing heat treatments. The NDT frequencies of the Navy high-tensile steel (HTS) and of ASTM A441 steels, as given in Fig. 8.36, illustrate the generally expected Gaussian distributions. Both steels are essentially equivalent with respect to chemical specification requirements, but the improved NDT of the HTS steel reflects the additional requirement for a normalizing heat treatment.

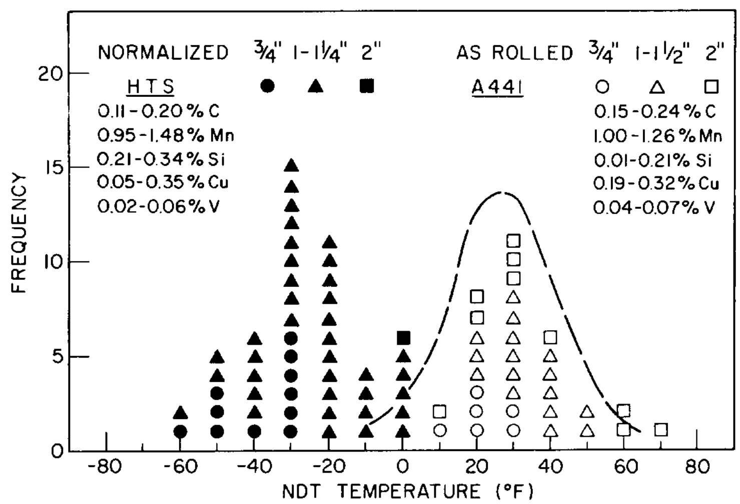



Fig. 8.36. Comparison of NDT Frequency Distribution Plot of As-Rolled ASTM A441 Steels with that of Normalized High-Tensile Steels. The illustrated curve represents the NDT frequency distribution plot for World War II ship steels (points not shown). (From ref. 61)

## 8.5.3 Neutron Embrittlement of Steel

The influence of nuclear radiation on the properties of the materials of construction of reactor secondary containment structures is not a design consideration because reactor primary systems are fully shielded for biological reasons. Moreover, secondary containment vessels, because of their physical displacement from the reactor core, normally would not experience significant radiation, even if biological shielding were not present. Nevertheless, since the primary purpose of the secondary containment system is to provide protection in the event of failure of the reactor primary system, a review of present knowledge of neutron embrittlement of steels is warranted as background information regarding the potential hazards associated with this phenomenon.

The conditions for catastrophic failure of structural steel components, such as the pressure vessel, and operational considerations for minimizing the potentialities of such failures were reviewed above. This section deals, in a summary manner, with experimental evidence of steel embrittlement by neutron irradiation and the influence of variable factors, such as neutron exposure and temperature, as determined from many test reactor experiments. The extent of observed embrittlement is sufficient to imply the need for restrictions and limitations on reactor pressurization during startup and shutdown for some reactors and to suggest further investigations employing power reactor facilities to verify both neutron embrittlement observations and the need for applying reactor operating restrictions.

### 8.5.3.1 Effects of Irradiation on Notch Ductility of Steels

The major deleterious effect of nuclear radiation on the carbon steels normally used for pressure vessel construction is the reduction of notch ductility, as evidenced by an increase in the ductile-to-brittle transition temperature (often defined as an increase in the nil ductility transition temperature, $\Delta$NDT), and by the reduction in the ability of the steel to withstand gross deformation or tearing above this temperature. Figure 8.37 shows the embrittlement that has been observed with irradiated ASTM Type A-302-B steel.[65] This phenomenon is attributed to bombardment by high-energy neutrons, which induce dislocations in the atomic structure of the steel.[66]

Experiments conducted in test reactors have shown, thus far, that three variables primarily determine the irradiation response of steels. These include the temperature of irradiation, the total neutron exposure (usually reported in terms of neutrons/cm$^2$ for neutrons with energy >1 Mev), and the composition or microstructure of the steel. Effects related to the neutron spectrum (distribution of neutrons by energy level in representative steel irradiation positions) and neutron dose rate (intensity of radiation) have not been found to be significant variables in test reactor experiments.[67,68] However, surveillance experiments are presently under way in both test and power reactors to examine further the influence of these exposure variables.[69-71]

1. Effect of Irradiation Temperature. The influence of increasingly higher exposure temperatures on the notch ductility of steels is manifested primarily by a reduction in irradiation embrittlement. The performance of steel in experimental assemblies having several progressively higher temperature zones has shown that this effect, termed "self annealing," does not become significant until the irradiation temperature exceeds approximately 450°F. At progressively higher irradiation temperatures, the

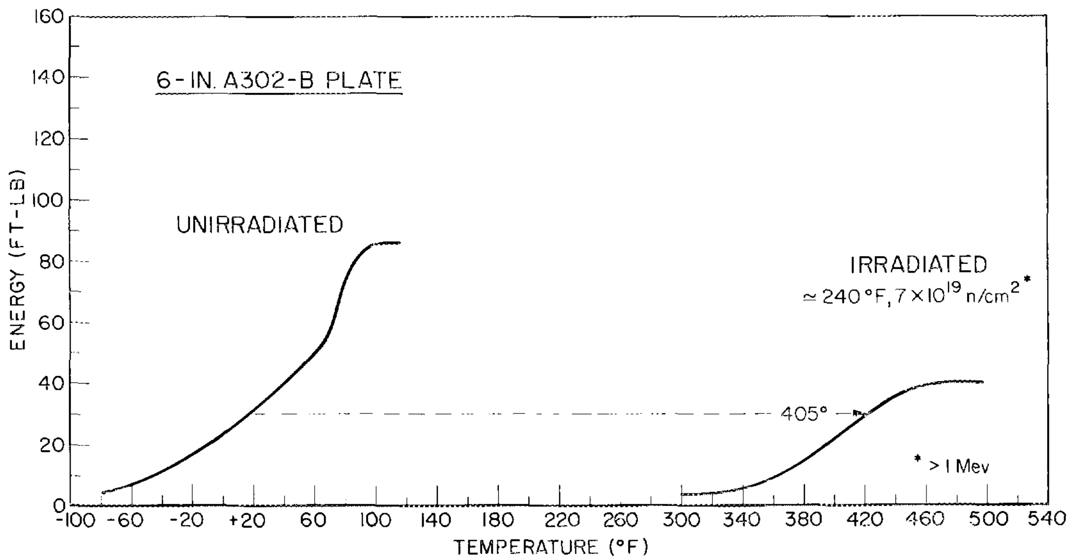



Fig. 8.37. Transition Temperature and Shear Energy Characteristics of Irradiated ASTM Type A302-B Steel. (From ref. 65)

annealing effect becomes more pronounced, with an equivalent reduction in apparent irradiation sensitivity, as demonstrated[72] for A-302B steel in Fig. 8.38. Since most pressurized-water reactors operate at temperatures in the range 400 to 600°F, various degrees of self annealing may

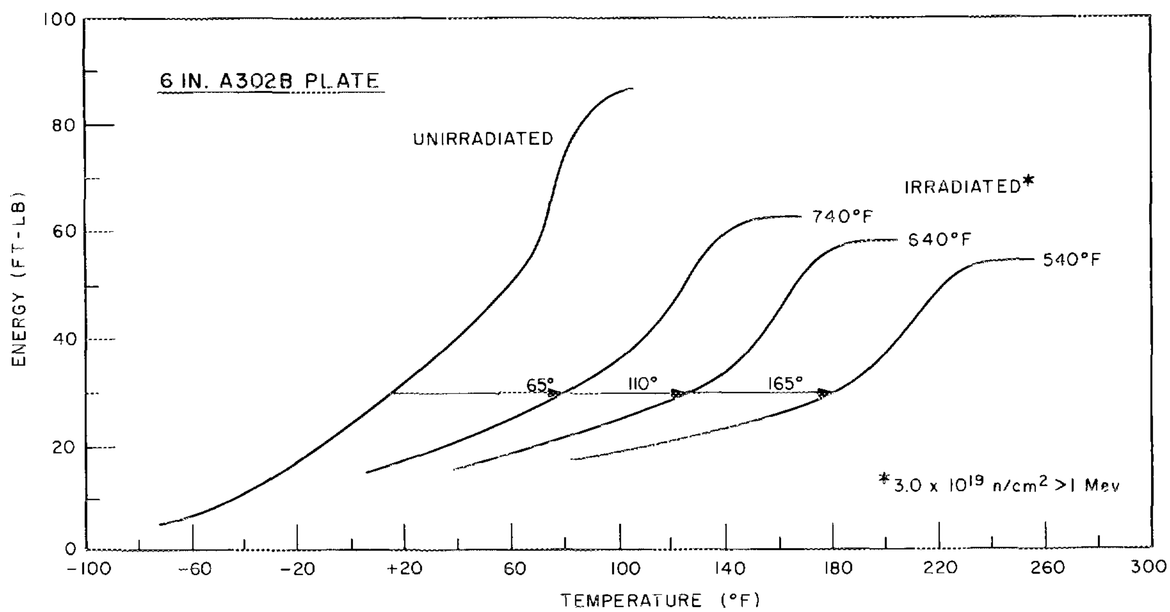



Fig. 8.38. Transition Temperature Shifts Resulting from Elevated-Temperature Irradiation of ASTM Type A302-B Steel. (From ref. 72)

be expected and must be considered in assessing the extent of neutron embrittlement. Experimental programs for more closely defining the temperature effects for various pressure vessel steels are now under way.

2. Effect of Neutron Dosage Accumulation. The continued neutron irradiation of steel under specific conditions results in a successively greater change in notch ductility properties, as demonstrated[73] in Fig. 8.39. The progressive embrittlement, however, is not linear. This is shown[74] for A-302B steel in Fig. 8.40. For exposures at temperatures less than 450°F, the first indications of change become apparent at a neutron dosage of approximately $1 \times 10^{17}$ neutrons/cm$^2$, although significant effects are not induced until the exposure reaches approximately $5 \times 10^{18}$ neutrons/cm$^2$. The increase in the NDT with higher neutron exposure is accompanied by a concurrent decrease in full shear energy absorption, as shown in Fig. 8.39. Under high neutron exposure, this property may be reduced to an undesirably low value from the standpoint of resistance to low-energy shear rupturing.

3. Effect of Material Composition and Heat Treatment. A compilation is presented in Fig. 8.41 of data obtained by one laboratory[75] on the transition temperature increase of several steels as a function of neutron exposure at temperatures less than 450°F. With this coordinate system, the points for individual materials irradiations fall within a relatively narrow trend band. A similar compilation[75] for elevated-temperature irradiations is shown in Fig. 8.42. On comparing these data to the trend band for irradiations, at less than 450°F, the progressive increase in the degree of self annealing with higher exposure temperatures is apparent.

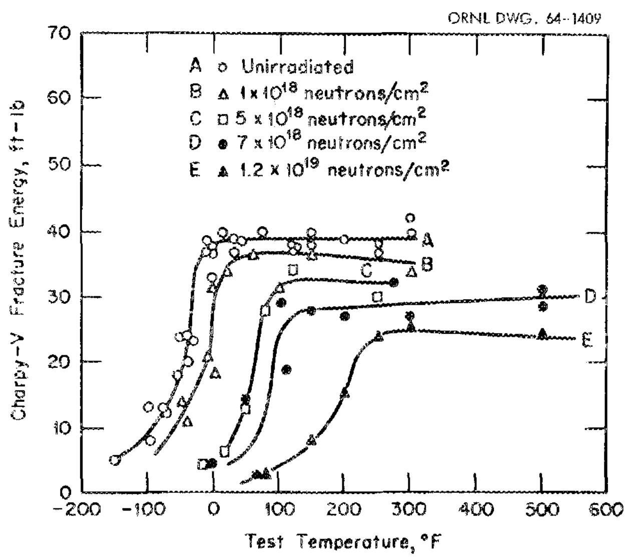



Fig. 8.39. Charpy V-Notch Impact Energy Curves for ASTM A-212 Grade B Unirradiated and Irradiated to Various Fast (>1 Mev) Neutron Doses at 175°F. (From ref. 73)

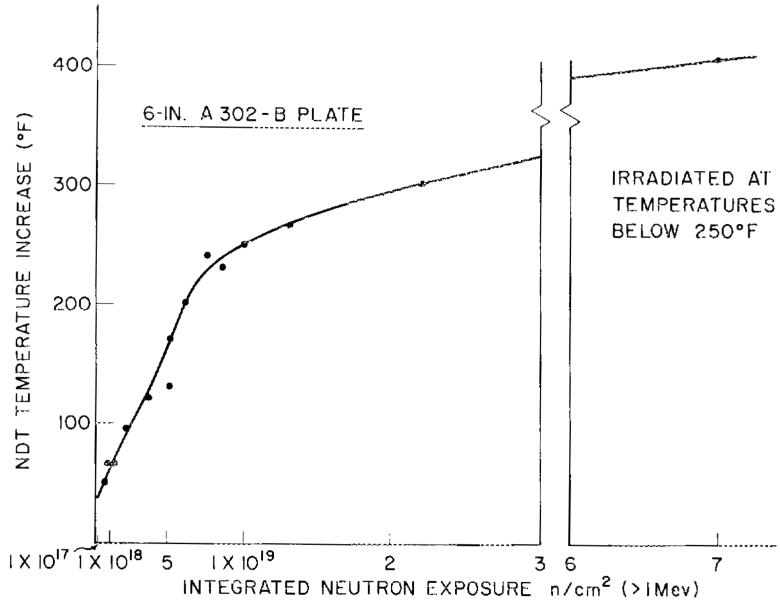



Fig. 8.40. Relationship of NDT Temperature Increase of ASTM Type A302-B Steel to Accumulated Neutron Dosage at Temperatures Below 450°F. (From ref. 74)

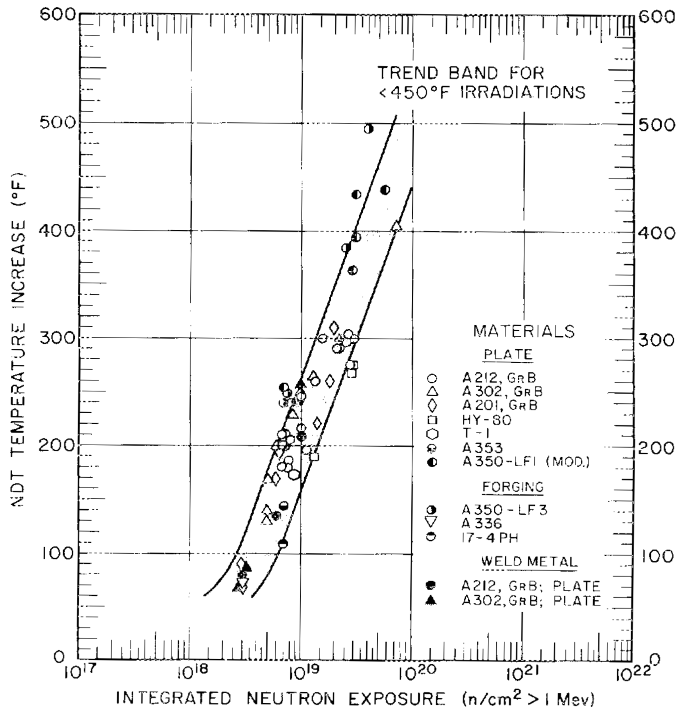



Fig. 8.41. Increases in the NDT Temperatures of Steels as a Result of Irradiation at Temperatures Below 450°F. (From ref. 75)

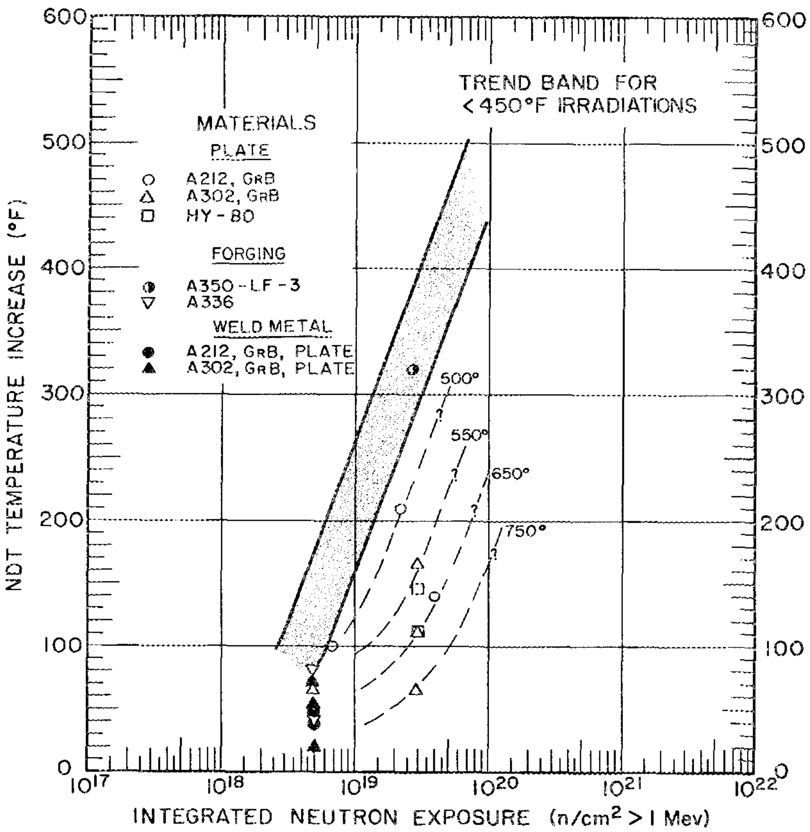



Fig. 8.42. Increases in the NDT Temperatures of Steels as a Result of Irradiation at Temperatures Above 450°F. Data points at $5 \times 10^{18}$ neutrons/cm$^2$ represent early irradiations in the Brookhaven Graphite Reactor (BGR) at 500 to 600°F. (From ref. 75)

Although a consistent relationship between transition-temperature increase with neutron exposure is evident in Figs. 8.41 and 8.42, the relative grouping of data points for individual steels suggests some variations in irradiation response between steels. Experiments for specifically comparing the relative irradiation sensitivity of select steels were performed recently. The data established that material composition and inherent microstructure are significant variables.[75,76] Data from one such experiment,[77] presented in Fig. 8.43, illustrate the importance of material selection for reactor systems. The variation in embrittlement in this experiment approaches a factor of 3 and thus suggests that materials sensitivity is as important as irradiation temperature in assessing the neutron embrittlement of a particular reactor vessel.

4. Major Changes in Exposure Conditions. The cumulative nature of neutron embrittlement at constant temperature is well established. However, nuclear power reactors are, in some cases, operated at various temperatures during a core life to conserve fuel or to extend fuel burnup. Since the neutron embrittlement of steel is a function of exposure temperature (over 450°F), the effects of raising and lowering operating

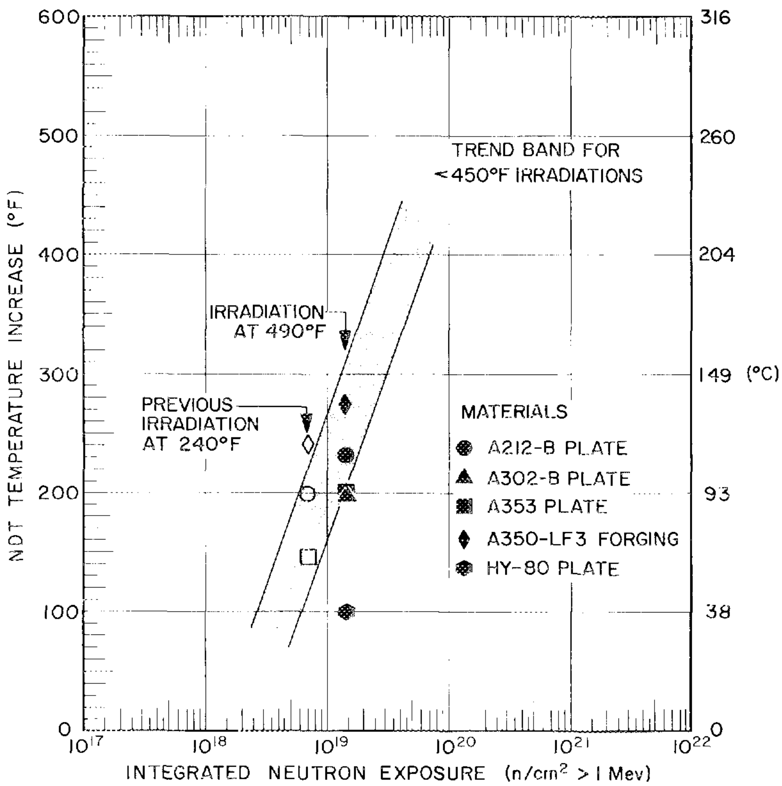



Fig. 8.43. NDT Temperature Increases of Five Steels Irradiated Simultaneously at 490°F. (From ref. 77)

temperatures must be considered in predicting the notch ductility properties of reactor components during service.

The effect of raising a pressure vessel service temperature after some period of operation is illustrated[78] in Fig. 8.44. Although a lower rate of progressive embrittlement (due to greater self annealing) is expected at the higher temperature, the data also demonstrate that some portion of the embrittlement incurred during the initial or low-temperature operating period was removed as well. Thus, the overall level of embrittlement, as well as the rate of continued embrittlement, is decreased by operating at a higher temperature.

The effect of a decrease in the pressure vessel operating temperature after a period of service is shown in Fig. 8.45. Here, adverse behavior is noted. The reduction in service temperature causes the steel to be embrittled at a higher rate, reflecting the lower self-annealing potential at the reduced temperature.

The data presented in Figs. 8.44 and 8.45 suggest that where service temperatures are varied, the more beneficial operating procedure entails an increase in component temperatures during core life and that the reverse procedure should be avoided if possible because of adverse embrittling effects during any period of lowered operating temperature.

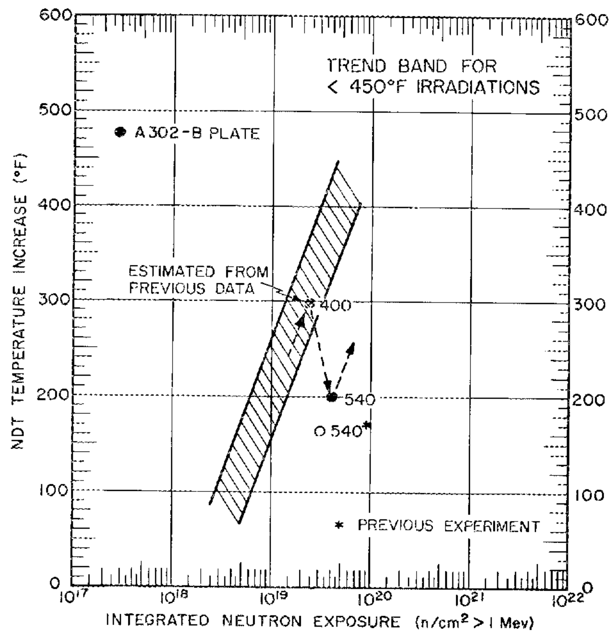



Fig. 8.44. NDT Temperature Increase of ASTM Type A302-B Steel as a Result of Irradiation in a Two-Phase Schedule. Phase one exposure conducted at 400°F. Phase two exposure performed at 540°F. (From ref. 78)

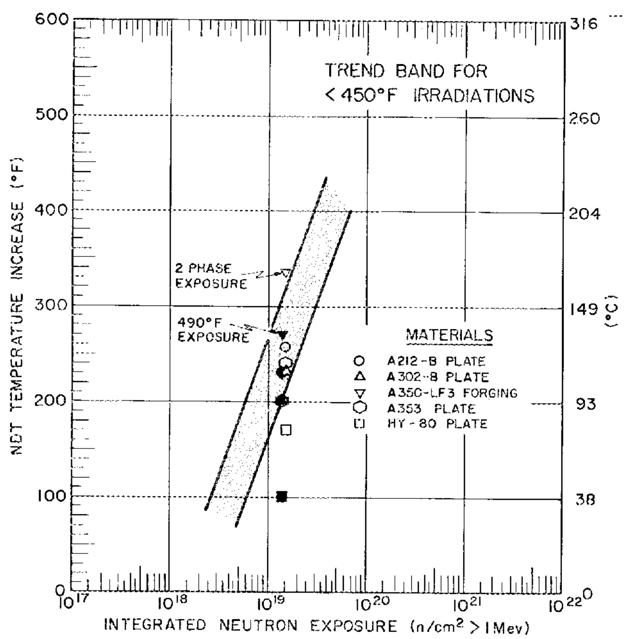



Fig. 8.45. NDT Temperature Increases of Five Steels After Irradiation Under Two Exposure Conditions. Two-phase exposure entailed irradiation at 490°F to $1.0 \times 10^{19}$ neutrons/cm$^2$ followed by additional exposure at 350°F. (From ref. 77)

8.5.3.2 Postirradiation Heat Treatment and Cyclic Irradiation and Annealing

The magnitude of the NDT increase developed in steels exposed at pressure vessel service temperatures (400–600°F) to moderate neutron dosages (1 to 5 × $10^{19}$ neutrons/cm$^2$ >1 Mev) has prompted investigation of the possibilities for recovering initial material properties. One method[72,75] proved partially effective for restoring notch ductility is postirradiation heat treatment (Fig. 8.46). The variables controlling heat-treatment response include the temperature during irradiation, the total neutron dosage received, material composition and microstructure, and the temperature and duration of heat treatment. In general, a greater amount of recovery is possible following exposure at lower temperatures and with lower neutron dosage accumulations. Similarly, greater recovery is achieved by high-temperature, long-term heat treatments than low-temperature, short-term annealing.

The promise of heat treatment as a method for notch ductility restoration has been explored further with investigations of the response of steels to cyclic irradiation and annealing treatments of the type envisioned for reactor application. The notch ductility behavior of one pressure vessel steel irradiated at 240°F, annealed at 700°F, and reirradiated, is shown in Fig. 8.47, which illustrates the possible benefits of intermediate heat treatment.[75] However, periodic annealing does not appear as highly beneficial when pressure vessel irradiation conditions are simulated and relatively low-temperature heat treatments are employed[78] (Fig. 8.48).

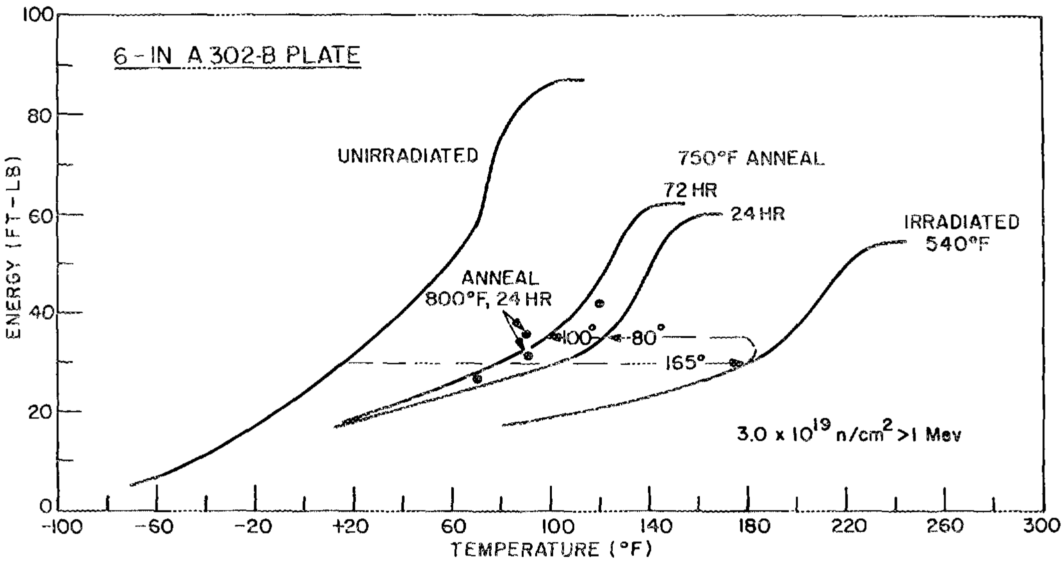



Fig. 8.46. Postirradiation Heat Treatment Response of ASTM Type A302-B Steel Exposed at 540°F. (From ref. 75)

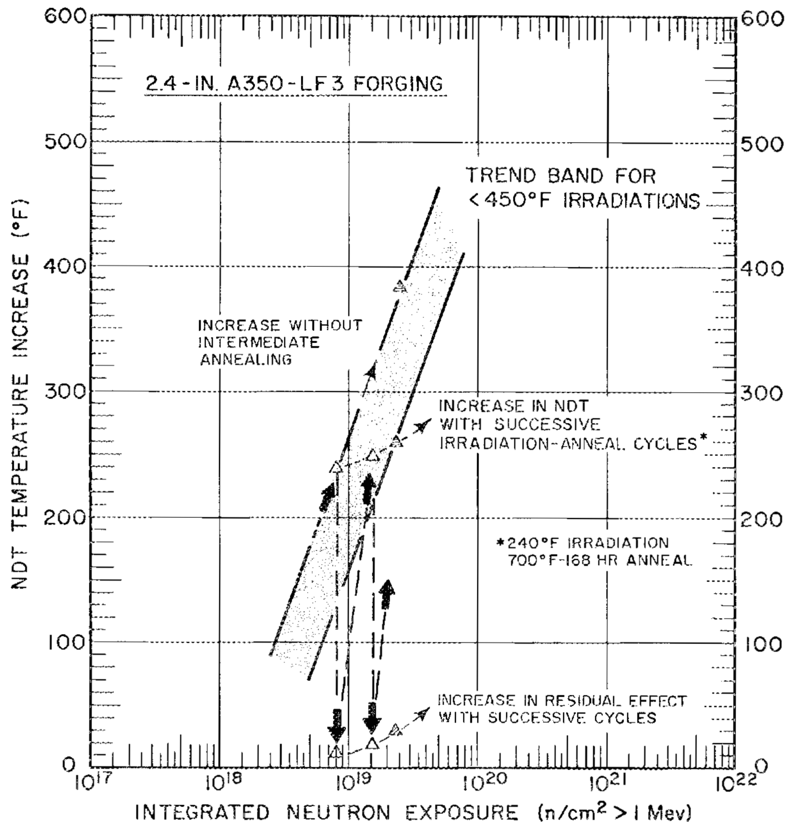



Fig. 8.47. NDT Temperature Behavior of ASTM Type A350-LF3 Forged Steel at Various Stages of Cyclic Irradiation and 700°F Annealing Treatments. (From ref. 75)

### 8.5.3.3 Effects of Irradiation on Tensile Properties

Investigations[79] of the influence of neutron radiation on the tensile properties of steels have shown that irradiation increases the yield stress and, to a lesser extent, the ultimate tensile strength (Table 8.14). In addition, irradiated steels exhibit less uniform elongation and reduction in area. Furthermore, the yield-point phenomenon that is characteristic of mild steels is often no longer apparent after irradiation and there is a decrease in work-hardening capacity. As with investigations on notch ductility behavior, studies have shown that, for a given steel, the temperature of irradiation and the neutron dosage received primarily govern the extent of change in tensile properties. Thus, it is to be expected that these changes are but another aspect of the same metallurgical consequence of irradiation that are the cause of notch brittleness discussed above. Because of the reduced notch ductility, as well as reduced ductility in tension, no advantage can be taken of the higher yield strength after irradiation. Much more data are required before the full significance of combined changes in tension test characteristics and notch ductility properties can be assessed.

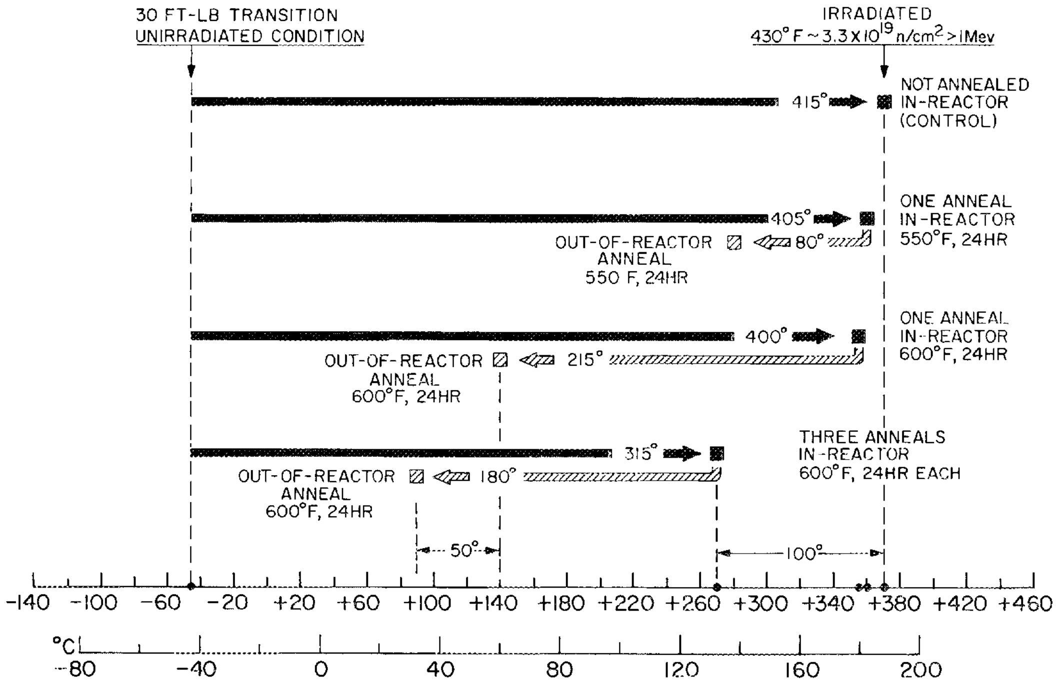



Fig. 8.48. Charpy-V Transition Temperature Behavior of ASTM Type A350-LF3 (Modified) Steel at Various Stages of Cyclic Irradiation and Annealing Treatments. The Charpy-V 30 ft-lb transition corresponds to the NDT temperature of this steel. (From ref. 78)

Table 8.14. Mechanical Strength Characteristics of Type A-212 Grade B Steel Before and After Irradiation[a]

| Property | Original Value | Value After Irradiation[b] | Change (%) |
|---|---|---|---|
| Yield strength | 40,000 psi | 76,000 psi | +36 |
| Rupture strength | 75,000 psi | 102,000 psi | +27 |
| Elongation in 1 in. | 25% | 9.0% | −64 |

[a]From ref. 79.

[b]Irradiation to a fast-neutron (>1 Mev) dose of $2 \times 10^{19}$ neutrons/cm$^2$.

### 8.5.3.4 Reactor Surveillance Programs

A recent survey by DiNunno and Holt of the U.S. Atomic Energy Commission[79] indicates that over half of the civilian reactors now in operation or in advanced stages of design and development have planned surveillance programs. These programs are specifically designed to permit the periodic "monitoring" of the mechanical properties of critical reactor components, such as the pressure vessel, and stem from unresolved questions as to the engineering significance of environmental differences between power reactors and test reactors. In addition, these programs provide a means for checking the accuracy of calculated values of neutron dose rate at specific locations within the reactor vessel. Every effort should be made to assure comprehensive surveillance programs in each new reactor system, since the data from these programs will supplement data from other sources. Furthermore, such programs[71] will add validity to the application of data from test reactor experiments to the power reactor situation, particularly where component stress and temperature conditions must be controlled during reactor startup and shutdown to avoid conditions favoring brittle fracture. A recommended program for reactor surveillance has been published by the American Society for Testing Materials (ASTM).[80]

To implement knowledge gained from reactor surveillance programs, investigations are being conducted on the pressure vessels and other steel components of decommissioned reactors, including the Army SL-1 and PM-2A reactors and the Organic Moderated Reactor Experiment. The examination of these components will be particularly significant in that the effects of service stresses during irradiation can be evaluated, along with the effects of nuclear service environmental conditions, for direct comparison with the body of data available from experimental programs conducted in test reactors. Data developed from reactor surveillance programs and from tests of reactor components taken out of service will, if in agreement with accelerated irradiated data, enhance the direct applicability of all data for predicting the embrittlement of reactor pressure vessels and other steel components.

## 8.5.4 Summary

This section has introduced in a summary manner the potential problem related to neutron embrittlement of the steels of the primary reactor vessel and the key factors determining the significance of such embrittlement. A complete discussion of neutron embrittlement has not been possible, but it is hoped that the problem of embrittlement and the implication for integrity of the secondary containment system has been demonstrated. More detailed descriptions of the problem of neutron embrittlement of steels are given in a number of excellent articles and reviews that discuss this phenomenon quite thoroughly.[81-86]

As has been pointed out earlier, the effects of neutron exposure on presently conceived containment vessels should be negligible. It is possible though that a situation might arise where such effects would be of significance. However, the likelihood of this is low since to obtain high fast-neutron exposure of a containment vessel there would also be a high thermal-neutron flux and a high gamma flux, such as to make the interior of the vessel uninhabitable and probably inaccessible for maintenance and repair. These factors tend to result in a low exposure level for any containment vessel. It is suggested that the primary considerations in the possibility of catastrophic brittle failure of a containment vessel will depend upon the initial material, design details, and the variables introduced by construction, welding, and postwelding heat treatment.

## 8.6 CONTAINMENT PROTECTION

### 8.6.1 Insulation

Steel containment vessels are often insulated. The usual reasons for insulation are:

1. To prevent excessive temperatures inside the containment vessel due to solar radiation.
2. To reduce heat loss in winter and reduce or eliminate condensation on interior surface,
3. To prevent the steel plate from approaching the nil ductility transition temperature, and
4. To minimize thermal expansion of the shell due to solar heating.

Most of these reasons do not apply to concrete containment structures, and insulation is not normally used for these structures. The controlled leakage refueling building at Hallam uses insulated metal panels to reduce winter heat loss.

One technical disadvantage of insulating a containment vessel is that it impedes the dissipation of heat following an accident. Other practical disadvantages are listed below:

1. Insulation systems thus far developed for steel containment vessels have limited service life — usually about five years.

2. Insulation in poor condition permits hidden corrosion to take place that otherwise could be easily noticed and corrected and makes leak detection difficult.

3. Many insulated vessels do not look attractive. This is often important in view of the size and prominence of the containment vessel in many installations.

4. Some insulation systems that are attractive initially may deteriorate to a less attractive condition in time.

Many types of insulation systems have been used on steel containment vessels. The Dresden and Big Rock Point containment vessels use a 3/8-in.-thick sprayed-on cork-filled asphalt mastic. This is applied to the primed steel plate without mesh or fasteners by means of a compressed air spray. Satisfactory application of the mastic depends on rigid inspection to avoid laminations and porosity. The color of the mastic material is black, and, to improve the appearance, a color coat was applied on top. However, because of the dark color and rough texture, it is difficult to find a satisfactory color coat. Further, the mastic must be allowed to cure before applying the color coat.

The HWCTR containment vessel is insulated with Styrofoam blocks coated with glass mesh reinforced mastic and paint. The blocks are held in place on the cylindrical shell with circumferential steel bands and an epoxy adhesive. On the dome the blocks are impaled on studs. The designers of the plant indicate that they would consider using a sprayed-on-polyurethane insulation if a similar vessel were to be constructed in the future.[25]

BONUS, in Puerto Rico, and Peach Bottom, in southeastern Pennsylvania, have uninsulated steel containment vessels. The reason for the lack of insulation on BONUS is the mild climate.[41] At Peach Bottom it was felt that internal condensation was acceptable and that the uninsulated shell could be used for natural dissipation of decay heat and thereby eliminate the need for a separate postincident cooling system. Yankee, which is located in a relatively cold climate, is not insulated because of the decay-heat release problem after a major loss of coolant accident. Table 8.15 lists the insulation materials used on various reactors.

Table 8.15. Insulation Materials Used on Various Reactor Containment Vessels

| Plant | Insulation |
|---|---|
| Dresden | Cork-filled mastic |
| Big Rock Point | Cork-filled mastic |
| HWCTR | Styrofoam blocks covered with glass mesh reinforced mastic and paint |
| BONUS | Uninsulated |
| Peach Bottom | Uninsulated |
| Humboldt Bay | Uninsulated |
| Hallam (underground vaults) | Uninsulated |
| EGCR | Uninsulated |
| Elk River | Foamglas and mastic coating |
| LACBWR | Foamglas and mastic coating |
| EBWR | Foamglas and mastic coating |

### 8.6.2 Paint

Steel containment vessels are always painted to provide corrosion protection. Often, a different type of paint is applied below grade than is used above grade. Corrosion below grade is difficult to discover and to correct, so a more expensive but better protective coating, such as metallic zinc, is sometimes used. Such coatings cannot be economically justified above ground. If cathodic protection is to be used, it is advantageous to employ a high dielectric coating, such as a coal-tar mastic.

Primers are usually applied in the shop to metal that has been pickled or blast cleaned according to the specifications of the Steel Structures Painting Council.[87] The primer is then touched up in the field after the vessel has been fabricated. Primers often used are red lead and zinc chromate. Where a steel containment vessel is to be exposed to the atmosphere and is not to be insulated, a metallic zinc paint, such as Amercoat Corp. "Dimetcote," may be used. Use of this paint requires that the plate be sandblasted to clean metal. Finish coats may be colored, as appearance requires, or may be aluminum to lessen absorption of heat.

The carbon steel liner plate in the suppression chamber of the Humboldt Bay plant was left unpainted. Since the suppression chamber is closed to the atmosphere, with very infrequent access, it seemed reasonable to conclude that equilibrium could be achieved with respect to corrosion. Any coating system would have had to have been periodically inspected and maintained if it were to retain its value. Allowing modest corrosion of the liner plate seemed a better solution. Therefore, the steel was not painted, except for a band at the water level. It was lightly sandblasted to remove accumulations of scale and rust that could be nuclei for corrosion.

The preparation and painting of concrete containment surfaces are no different from the techniques used on concrete surfaces in nonnuclear construction. However, surface coats may be of lower porosity to reduce absorption and simplify decontamination. Painting of interior surfaces should be done to provide for the deposition and retention of fission products, as well as their subsequent decontamination.

### 8.6.3 Cathodic Protection

Cathodic protection is often employed to retard corrosion of inaccessible surfaces of steel containment vessels, such as the areas below grade. However, when incorrectly used, the imposed electrical currents can add to natural currents and can hasten corrosion. A successful installation depends on many site characteristics, including soil corrosivity, resistivity, and the nature and location of any existing electric grounding mats and structures. Often a survey by a consultant in this field is necessary to determine whether cathodic protection is desirable and how it should be applied. In order to reduce current requirements, a high dielectric coating is applied to the steel plate below grade.

The cathodic protection systems used at Dresden and at Big Rock Point contain many buried sacrificial electrodes around the sphere. Several reference electrodes were also installed under the spheres to enable checking the polarity between the sphere and the surrounding soil. Cathodic protection was not used at Humboldt Bay Unit No. 3, since there were solid electric grounding connections to the existing units which were not cathodically protected. Uhlig's book "Corrosion and Anti-Corrosives"[88] is a useful source for further information on this subject.

## 8.7 INSPECTION AND CONTROL

Careful inspection and testing must be performed in accordance with standard procedures to ensure that materials properties meet or exceed those specified. This section outlines the procedures followed for inspecting and testing concrete, steel plates and components, welding and field fabrication, and coatings. It also refers to the detailed test procedures normally specified for these materials.

### 8.7.1 Concrete

Inspection and control of concrete construction for reactor containment is usually performed by a testing organization and by the constructor's field inspector. The testing organization will examine the local supply of aggregate and cement and will design the mix, supervise batching, make field tests, and take and test sample cylinders. Often the design engineer will specify that small quantities of additives such as accelerators, retarders, or densifiers be mixed in the concrete during batching to improve the placing and durability and to reduce the shrinkage of the finished concrete.

The field tests and checks usually carried out by the inspector are the following:

1. On large jobs, concrete and concrete aggregate are continuously and routinely checked to assure uniformity.
2. On all jobs, slump should be checked continuously as a measure of consistency, workability, and a rough indication of the water-to-cement ratio.
3. During very hot weather or freezing weather, the temperature of concrete is taken as an indication that the proper amount of heated water or ice has been added so that the concrete will set properly.
4. Forms and reinforcing bars are inspected to assure that they will not subside or deflect under the weight of the concrete. Placement, vibration, and curing of the concrete are observed and inspected to assure the formation of hard, dense concrete that is free of voids and honeycombs.
5. Samples of concrete are taken, cast into 6- by 12-in. cylinders, and stored in a controlled environment prior to shipping to a testing laboratory. Occasionally, the testing laboratory is located at the job-site. At the testing laboratory, concrete cylinders are tested for compressive strength 28 days after pouring and often, in addition, after 7 days to exercise better control of mix design. If the concrete is used as shielding, the cylinders are also weighed to check density.

Much of the foregoing inspection is dependent upon individual judgment, and from this standpoint concrete construction is not as amenable as all-steel containment vessels to rigorous quality control. Assurance of adequate reliability demands constant and rigorous surveillance of all operations and careful recording of concrete mix and placement.

### 8.7.2 Steel

The quality of steel is controlled principally by inspection and tests at the mill. The purchaser of the steel obtains a certified statement from the mill that gives the results of tests, minimum strengths, and chemical content. Mill tests cover some or all of the following properties:

1. Yield Point (minimum). For carbon steel, the yield point is the lowest stress at which strain increases without an increase in stress.

2. Tensile Strength. The maximum or ultimate tensile stress that the material can sustain without failure is the tensile strength.

3. Elongation. The percentage increase in length (normally of a 2-in. or an 8-in. sample) of the material at the point of ultimate stress in tension is a measure of the ductility of the material.

4. Bending. A test specimen is bent through 180° on a specified radius. The specimen must not show any cracking.

5. Ladle Analysis. An analysis of each melt is made to determine the percentage of constituent elements, such as carbon, manganese, phosphorus, sulphur, copper and silicon.

6. Charpy Keyhole, U-Notch, or V-Notch Test. An impact test is conducted usually at a low, controlled temperature as a measure of the ductility toughness and brittle failure resisting properties of the steel. (See ASTM Spec. A-370.)

7. Homogeneity Test (for Firebox quality plates). The homogeneity test is made to disclose seams, laminations, etc. caused by improper rolling or inclusions of foreign matter or gas in the material. The test is made by inspecting a fractured surface of the plate. (See ASTM Spec. A-20.)

Other tests often performed at the mill or performed after the steel is received by the fabricator, are as follows:

1. A check analysis is sometimes made by the purchaser from finished material representing each melt to ensure that the percentages of the most important constituent elements are within the specified ranges.

2. Plates are visually checked for thickness and possible laminations.

3. In the field, the inspector will check principally for pitting of the plates.

U.S. practice does not require mechanical tests of plates or welds after fabrication or erection. Tests for reinforcing bars are not as extensive as those for containment plate. Nevertheless, reinforcing bar fabricators must provide mill test reports for materials provided from each heat. Field inspectors must check tags and symbols on the bar indicating size, origin, and, occasionally, type of steel.

### 8.7.3 Welding and Other Field Fabrication Operations

Welding operations must be performed by qualified welders in accordance with an approved procedure and be inspected by a qualified inspector. Details of these qualification requirements are outlined in Section VIII of the ASME Boiler and Pressure Vessel Code and applicable Code cases. Tests on containment vessel welds include radiography (usually full), magnetic particle, dye penetrant (usually used on joints where radiography is impractical), and soap-bubble tests.

In addition, there are special quality-control requirements for stress relieving of welds and for penetration nozzles, as outlined in Section VIII and incorporated by Section III of the ASME Code. The most significant feature of the quality-control procedures required for containment vessels is the magnitude of the job; containment vessels can easily have over 3 miles of welded seams, every inch of which must be radiographed and soap-bubble tested, in addition to being subjected to leak and strength tests, as discussed in Chapter 10.

Since metal linings are not covered by any vessel code, the usual practice is to completely spell out the material and testing requirements in the specifications for fabrication and erection of the metal liner. Ductility is not as critical for a liner as it is for a pressure vessel. However, it is still necessary to require mill tests to substantiate weldability and material composition of the liner plate. Current practice has also required welder qualification and testing of welds by spot radiography (10%). Full-coverage leak testing by vacuum box, halide, or other means has also been conducted to ensure leak tightness of the welded joints. Where the lining is welded to inserts and radiography is not possible, magnetic particle inspection can be performed. In the construction industry, inspection of reinforcing-bar welds is usually limited to visual inspection for cracks, inadequate size, and other visible defects, but it also includes magnetic particle inspection and radiography for highly critical applications or joints of suspected quality. Inasmuch as containment vessels may be considered a critical application, inspection of welded reinforcement bar should be done by visual, magnetic particle, and radiography methods.

### 8.7.4 Nonmetallic Coatings

For nonmetallic coatings, quality control is required for both the coating and the surface to be coated. All manufacturers of plastic coatings require special care in the preparation of the concrete surface to ensure the proper bond of the coating. The preparation usually includes an acid wash and filling of all surface holes over 1/8 in. with an inert material or material compatible with the coating. Because of the critical mixing and application requirements of the coating materials, the job requires special equipment and a background of experience. To ensure the quality of their product, manufacturers of most coating materials maintain close field liaison with the subcontractors applying them and are prepared to furnish lists of recommended subcontractors.

When coatings are used as leakage barriers, it is also important to maintain close inspection on all details that provide sealing at construction joints, changes of material, penetrations, and inserts. Tests for wet-film thickness, dry-film thickness, and pin holes should probably be conducted to ensure adequate coating.

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# 9. CONTAINMENT ACCESSORIES

R. F. Griffin*     G. H. Dyer*
H. W. Wahl*

## 9.1 INTRODUCTION

It would be a relatively simple matter to provide a leaktight containment structure around a reactor were it not for the fact that the reactor system requires electric power, a means of removing heat, occasional servicing, and a host of other utilities and services that require passage of pipes, wires, people, and equipment through the containment boundary. Containment integrity must still be maintained, however, over a wide range of both normal and abnormal operating conditions, including accident conditions. Thus a complete containment system must include numerous accessories, as well as the basic containment structure, in order to perform its required functions properly.

The requirements for accessories vary widely for different containment systems, depending, among other things, on reactor type, size, and operational requirements; on the possible consequences of a release of radioactivity to the environment, that is, on the site location and meteorological conditions; and on the type of containment used. For example, the number and size of air locks required for access to the containment system, and even whether or not air locks are required at all, depend on the degree of access required during reactor operation, during other times of potential hazard, and following an accident. The number, types, and operating characteristics of isolation valves required for pipes that penetrate the containment boundary depend on the type of pipe, the fluid carried, and the consequences of a pipe failure. A primary steam line from a direct-cycle reactor, for example, requires more isolation valving protection where it penetrates the containment boundary than does a small service water line. A pressure control system may form the entire basis of the containment design or it may merely serve to reduce the consequences of an accident below those for which the plant was designed. The same may be true of filter systems. In spite of these differences, however, some generalizations can be made about these accessories and some designs or design principles that are applicable to many different situations can be discussed.

This chapter describes numerous examples of access, piping, fuel transfer, and electrical penetrations and presents information on certain specific designs of each of these general types of penetrations. Such penetrations are frequently designed to facilitate either the continuous or periodic leak tests that may be desired or required to complement or take the place of integrated containment leak tests. Ventilation and pressure-control systems and fission-product removal systems are discussed in more general terms. Instrumentation and power requirements for operation of the containment system are discussed briefly.

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*Bechtel Corporation, San Francisco, California.

## 9.2 ACCESS PENETRATIONS

### 9.2.1 General

Access penetrations include both the openings in the containment structure that allow personnel and equipment to pass through the containment boundary and the closures that seal these openings to maintain containment integrity whenever it is required. Containment integrity must be maintained when substantial amounts of radioactivity could be released to the containment. Therefore air locks that provide continuous containment, even when being used for access, must be installed if access to the containment vessel is required during such periods. For water-cooled reactors, it is usually assumed that rapid releases of radioactive material can occur only when the reactor is operating or when the primary system is pressurized. The possibility of primary system rupture and subsequent rapid fuel meltdown occurring under cold shutdown conditions is normally not considered credible. However, meltdown of a fuel element inadvertently left without sufficient cooling for extended periods during dry refueling operations may be considered credible. Therefore, single-barrier access doors or ports that can be closed in a relatively short time interval may be used under these cold shutdown conditions. The quantity of radioactive materials that could credibly be released from a particular reactor plant under various conditions must be evaluated in each case to determine whether double-barrier lock systems are required or whether easily closed single-barrier doors may be used.

The following types of access penetrations are frequently used in conventional steel-shell types of containment vessels:

1. Large bolted openings are used for equipment transfer only during plant shutdown. An opening of this type, similar to the equipment doors used at the Dresden and Big Rock Point plants, is shown in Fig. 9.1.

2. Large equipment locks are used for transfer of equipment or fuel casks during plant operations. A typical equipment lock similar to those used at the Dresden, Big Rock Point, and Peach Bottom plants is shown in Fig. 9.2.

3. Personnel locks are used for normal passage of personnel in and out of the containment vessel during operation. A spherical air lock for use primarily in spherical containment vessels is illustrated in Fig. 9.3. This type of lock is used at Dresden. Cylindrical locks are also commonly used in both cylindrical and spherical containment vessels.

4. Escape locks are used for emergency exit from the containment vessel. An escape lock such as that used at Dresden and Big Rock Point is shown in Fig. 9.4.

Various other types of doors with suitable seals have been used in low-leakage buildings used for containment. Some of these are discussed in Section 9.2.4.

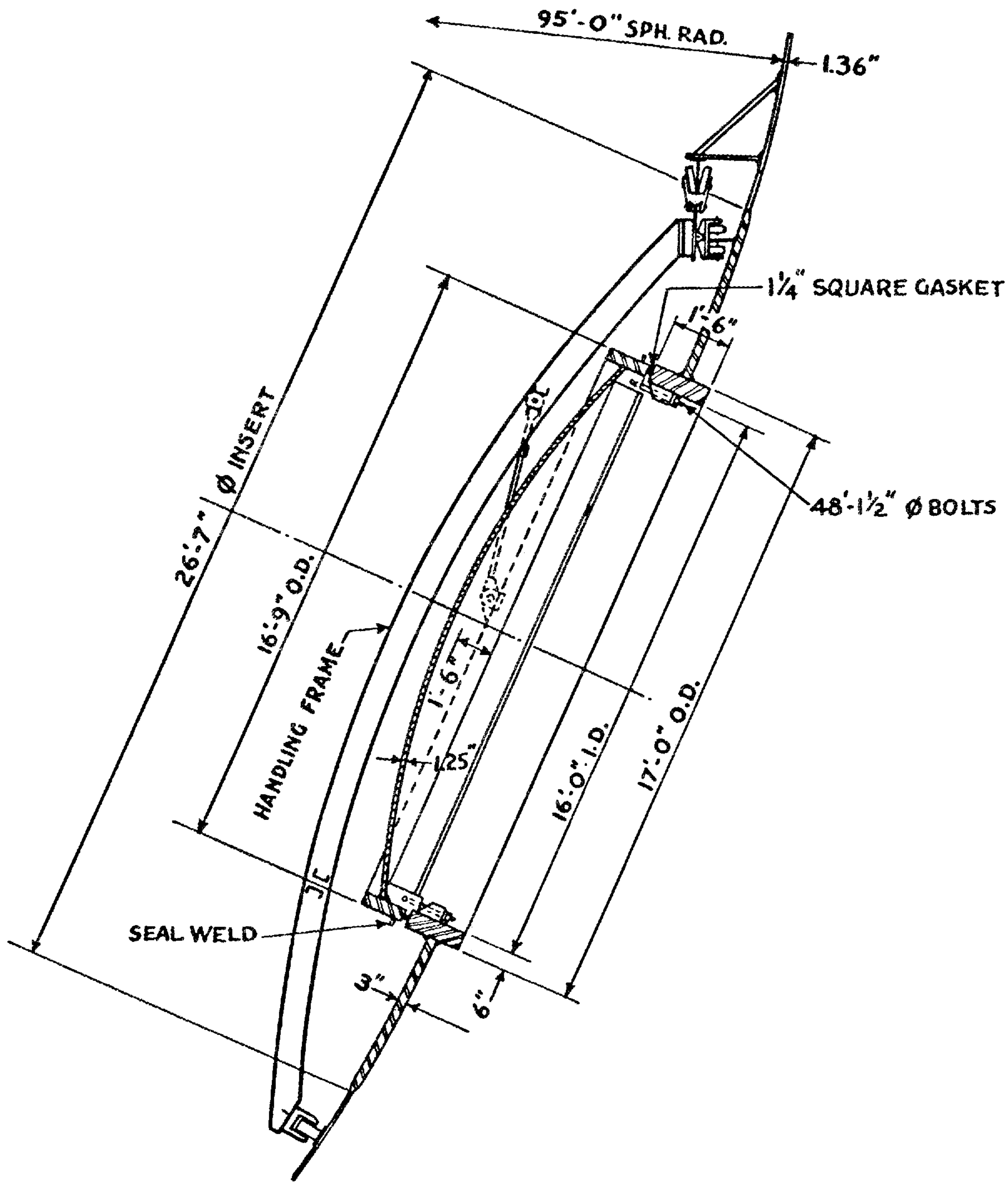



Fig. 9.1. Section Through 16-ft-diam Bolted Opening. (From Chicago Bridge & Iron Company)

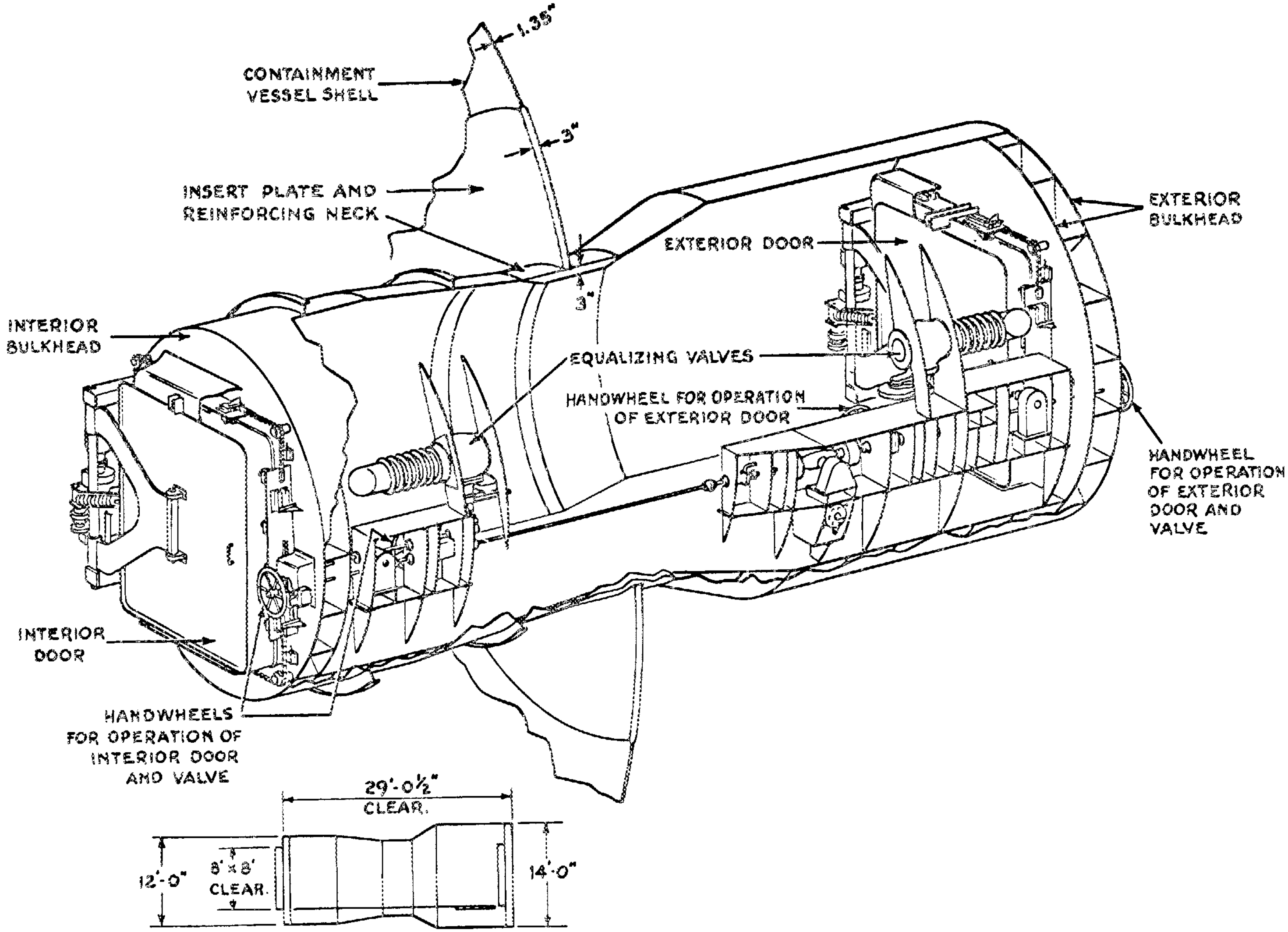



Fig. 9.2. Equipment Lock with 8-ft by 8-ft Doors. (From Chicago Bridge & Iron Company)

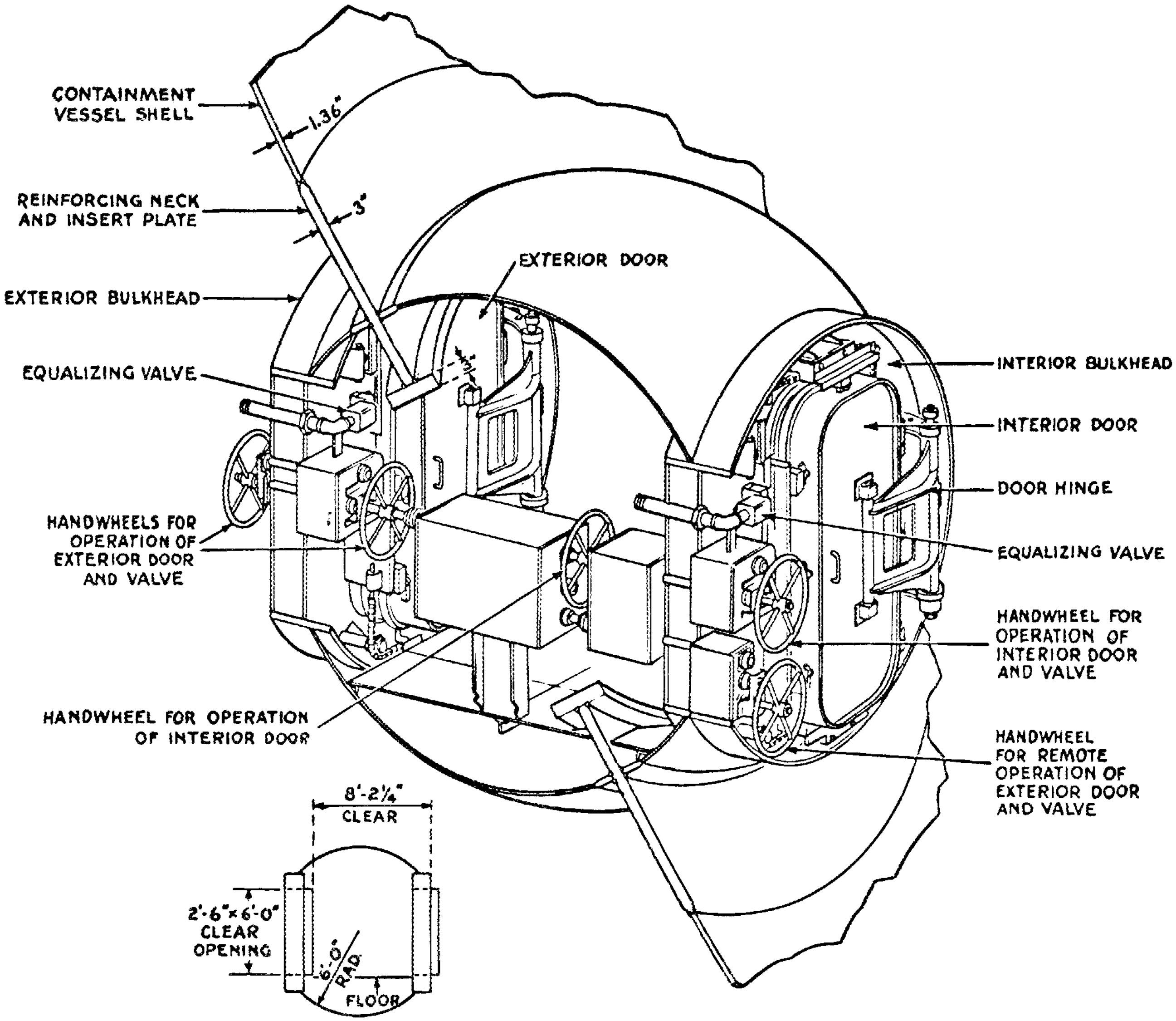



Fig. 9.3. Spherical Personnel Lock with 2-ft 6-in. by 6-ft Doors. (From Chicago Bridge & Iron Company)

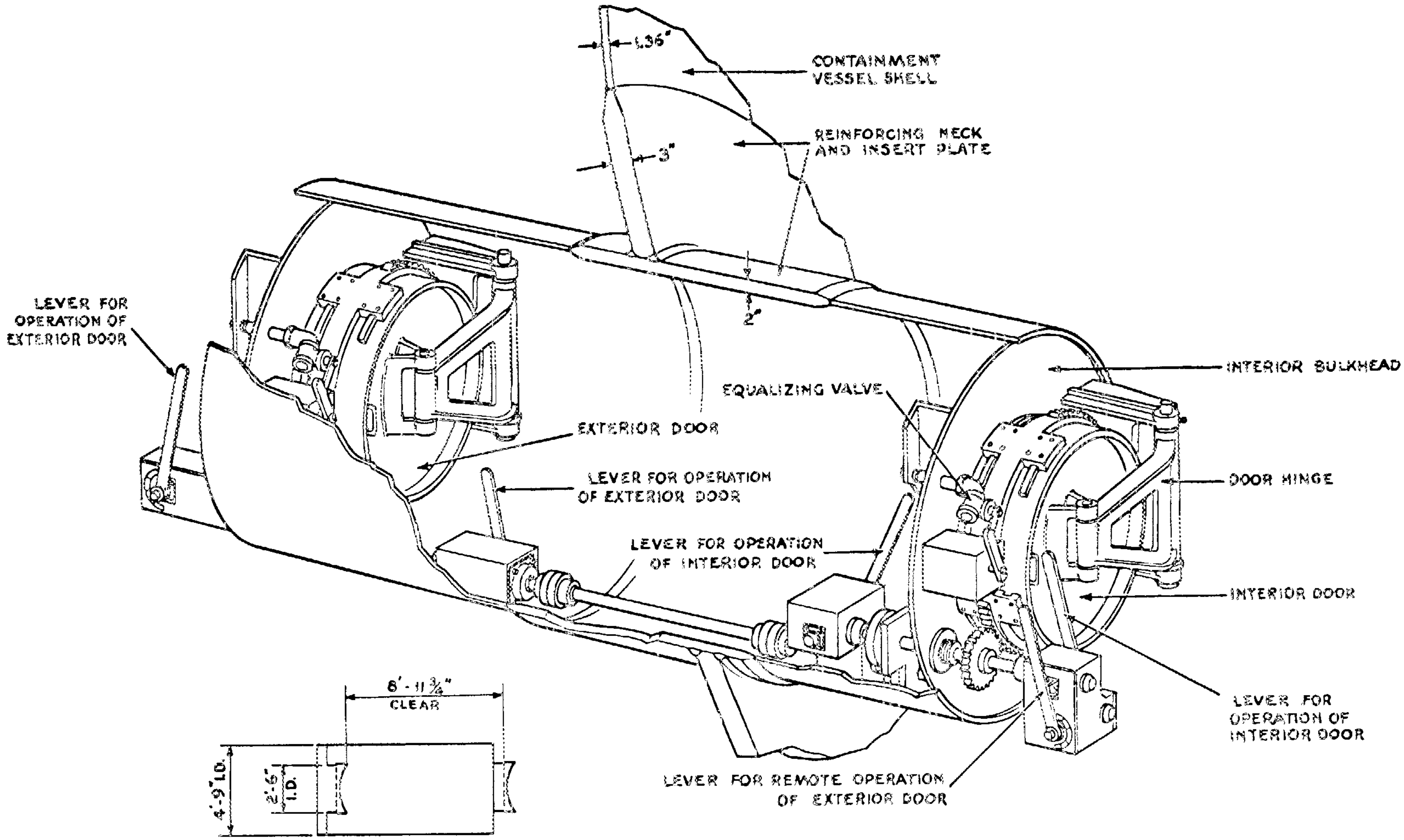



Fig. 9.4. Escape Lock with 2-ft 6-in. Doors. (From Chicago Bridge & Iron Company)

## 9.2.2 Air Locks

The minimum requirements for personnel air locks are listed below:

1. The lock structure and both doors must be designed and constructed for the same pressure as the containment vessel.
2. Equalizing valves must be provided to eliminate differential pressure across the doors so that they can be opened. The equalizing rate should be as fast as possible without causing personnel discomfort.
3. The doors must be interlocked to prevent both being open simultaneously. The equalizing valves must be connected to the operating mechanism of the doors in such a way that one door must be completely locked and its accompanying equalizing valve closed before the opposite door can be opened.
4. A mechanical pointer, windows, or indicating lights must be provided to indicate the position of the interlock, equalizing valve, and far door.
5. An interior lighting system, with an emergency power supply, must be available.
6. There must be an emergency communications system.

In addition to the above, the following provisions will also occasionally be made:

1. There may be a bypass on the door interlock mechanism to allow doors to be left open under proper authority during plant shutdown.
2. There may be a means of closing the far door in order that the near door may be opened.

Power operation of personnel access locks is optional, since the weight of the doors is usually such that manual operation is feasible and, in many cases, preferable. If power operation is used, alternate manual means must be provided to ensure that the lock can be operated in the event of a power failure. Manual operation avoids the higher cost and maintenance associated with power-operated doors and results in more trouble-free operation, but it may be inconvenient if frequent access to the containment vessel is required. The time required for manual operation is influenced by the design pressure, the size and type of the doors, and whether the doors are pressure-seated. Power operation (electric or air) is used for large or heavy doors. Escape-lock doors are always manually operated because they are small and because of the nature of the service required.

In addition to the above requirements for personnel locks, equipment locks have the following special requirements:

1. They must be of sufficient size to accommodate the largest equipment to be transferred during plant operation.
2. Special floor systems and floor or overhead tracks that can be removed must be provided to allow opening and closing of doors.
3. Special reinforcement of the lock and the vessel around the opening is necessary because of its large size.
4. Stiffening of the shell or special supports are required to carry the moving load of the equipment inside the lock.

Power operation is normal for equipment locks because of the weight of the doors, the force required to seat the locking wedges, and the operation of large equalizing valves.

In spite of the significant cost of equipment locks (see Chap. 11) and the savings to be realized by using a single equipment door, locks are frequently used for one or more of the following reasons:

1. In large plants with multiple coolant or recirculation loops, equipment locks are provided so that the replacement or repair of pumps and equipment from an isolated loop may be accomplished while the plant is operating or is at elevated temperature or pressure conditions.
2. When the spent fuel storage pit is inside the containment vessel and shipment of spent fuel casks may be too frequent or inconvenient to be done only during cold shutdown conditions, equipment locks are used so that shipments can be made during operation.
3. Equipment locks may be provided for access during operation for maintenance equipment, such as a fork lift truck or a disposal cask for radioactive materials, that is too large for a personnel lock.

Manufacturers are now offering complete lines of personnel and equipment air locks in various size ranges and pressure ratings. These locks are standardized, guaranteed, and available for service up to 50 psig from some equipment manufacturers as catalog items. They can be furnished completely inspected and stamped with the applicable ASME Code symbol and can be installed in any vessel.

### 9.2.3 Air-Lock Accessories and Details

#### 9.2.3.1 Locking Devices

A positive locking device is required on doors to resist the design pressure loads and maintain adequate gasket seat pressure. For the large access doors used only during shutdown, this locking function is accomplished by bolting. On air-lock doors, however, the locking mechanism must be quick acting. Multiple dogs operated simultaneously or a rotating collar engaging fixed wedges, as shown in Fig. 9.5, can be used for this purpose. The latter design was used for the air locks in the Big Rock Point plant.

Doors are usually designed to swing inward so that any internal pressure assists in seating. Equipment locks are sometimes exceptions to this rule, however, since a door opening into the lock would require the lock to be longer and would require a removable floor in part of the lock. If the containment vessel internal pressure tends to seat the door, the locking mechanism is not usually designed to hold more than a few pounds pressure in the lock with atmospheric pressure on the opposite side of the door. Where the lock must be pressurized to a higher pressure for testing purposes, temporary hold-down bolts are usually installed to prevent the inner door from opening.

All hinges should be provided with means for three-dimensional adjustment to assist in proper seating. The hinges should be capable of adjustment, independently of the door, in order to compensate, if necessary, for any deviations of the lock from a horizontal plane.

In low-pressure containment systems, regular ship-type quick-acting doors may be utilized. These are discussed in Section 9.2.4. The higher design pressures of conventional containment systems require that the

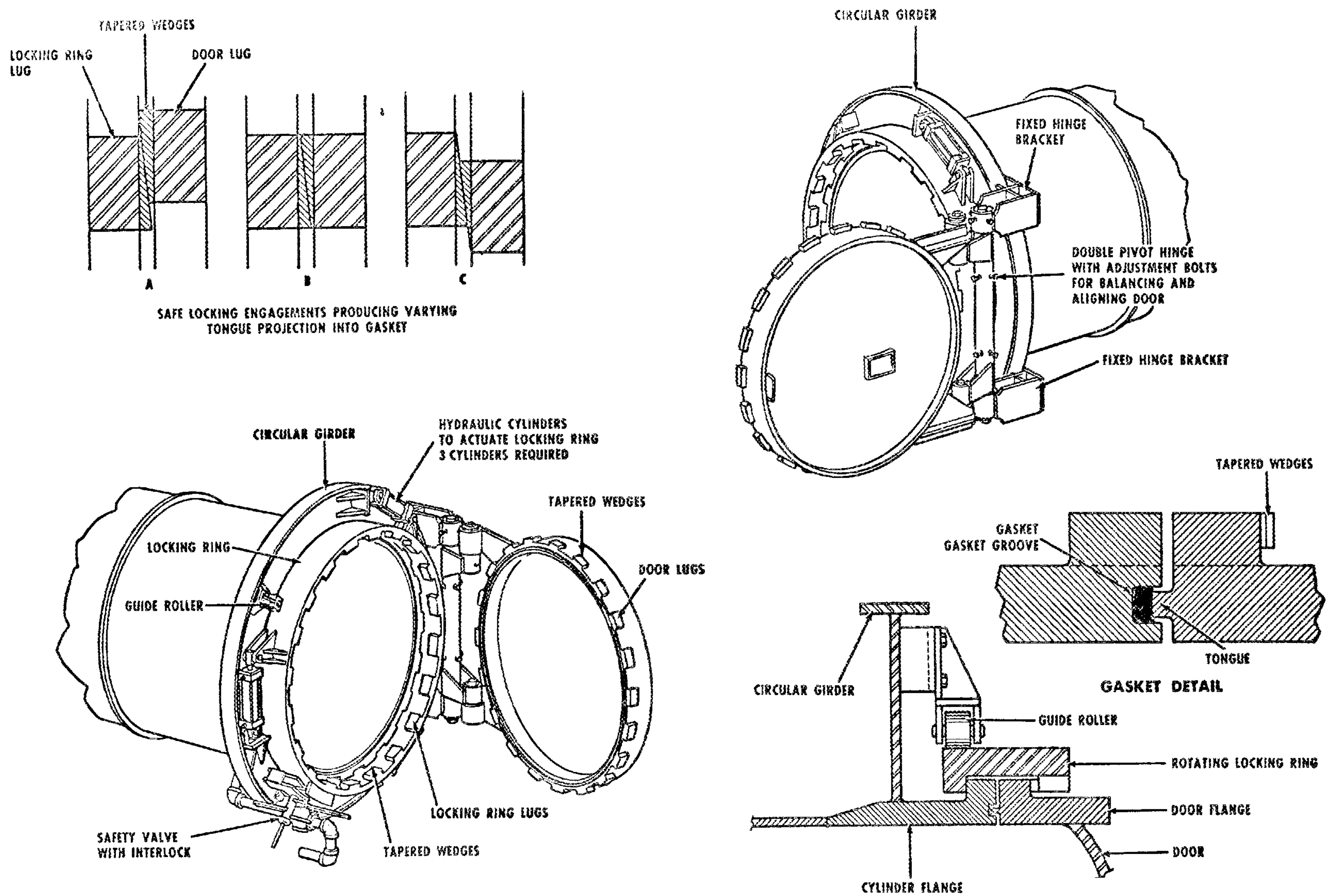



Fig. 9.5. Breech-Lock Type of Quick-Opening Door. (From Chicago Bridge & Iron Company)

door assembly and seat be sufficiently stiffened and aligned properly to prevent any warping under load.

### 9.2.3.2 Seals and Gaskets

Door perimeters are usually sealed with soft elastomer gaskets or inflatable seals. Typical seals are shown in Figs. 9.6 (ref. 1) and 9.7 (ref. 2). Gasket seals are used on both bolted and quick acting doors, but inflatable seals are not normally used for bolted doors. Single doors or bolted covers may have double gaskets to permit testing the seals without pressurizing the entire containment vessel. Seals must also be provided on gear shafts that penetrate the doors.

Softness of the rubber used in gaskets ranges from Durometer 40 to 60. The gasket may be contained in a machined groove, with sealing accomplished by a rounded bead projecting into the elastomer. This type of gasket is good for any pressure encountered in containment design. Inflatable elastomer seals are somewhat more expensive than gaskets but

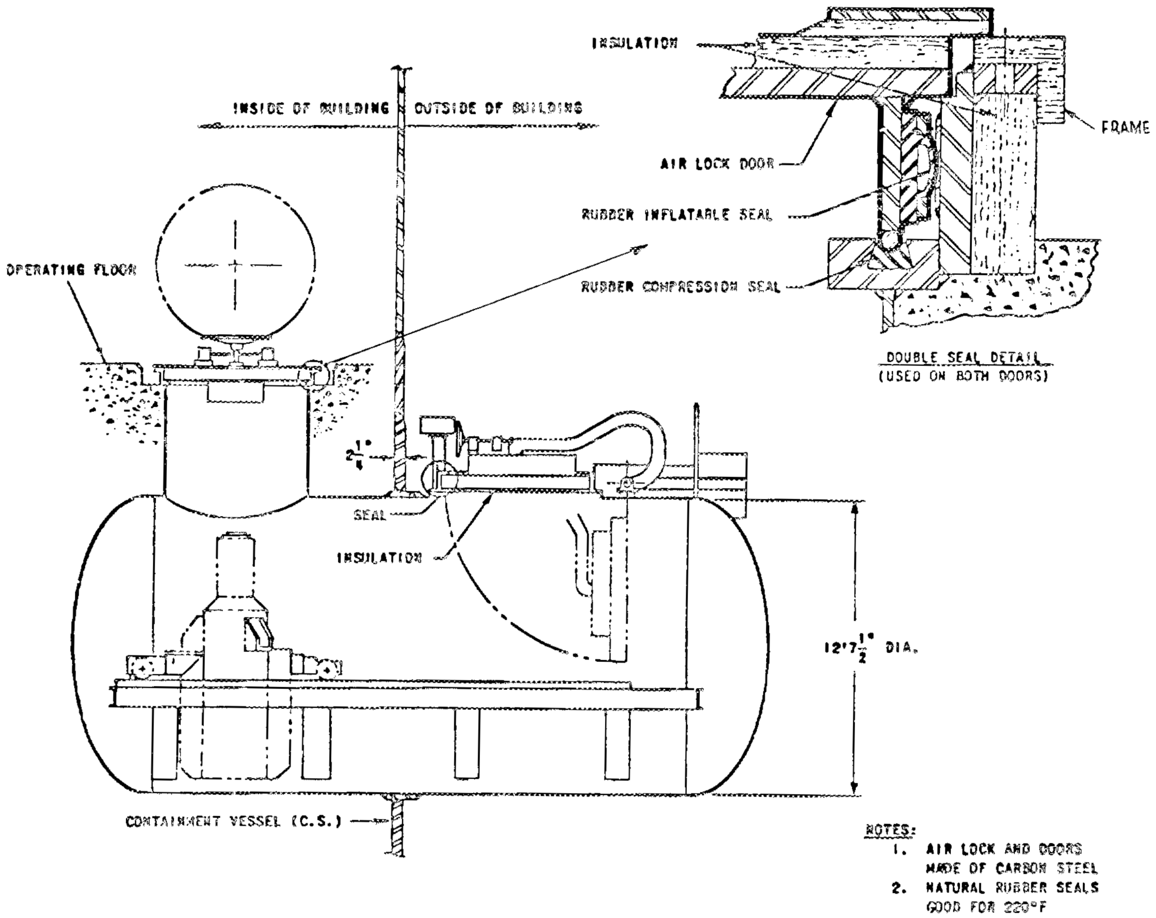
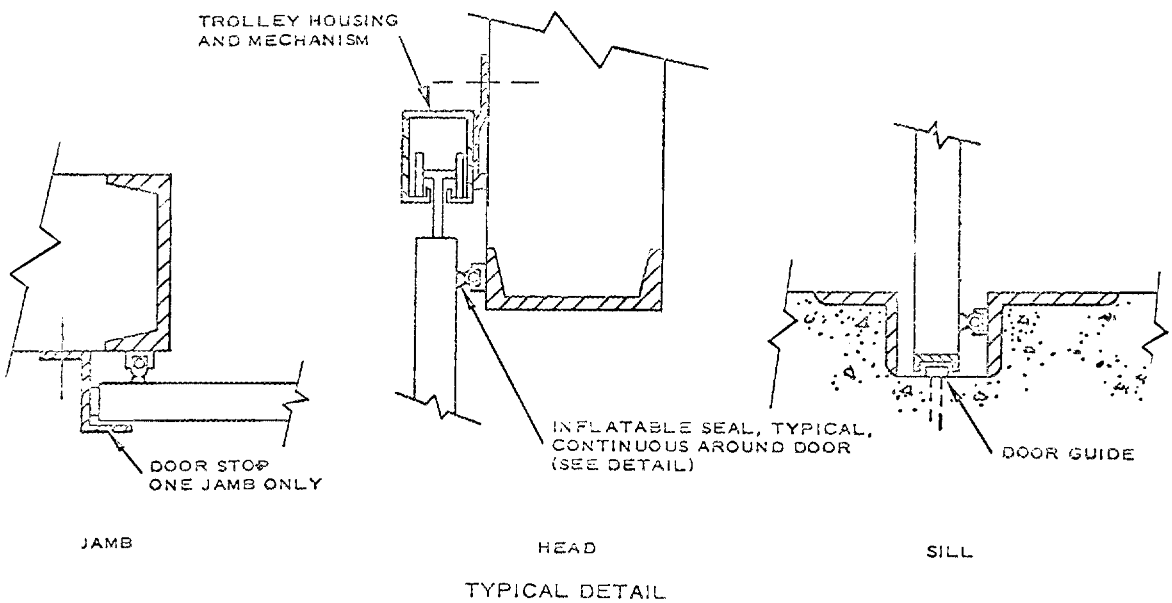



Fig. 9.6. EBR-II Equipment Air Lock. (From ref. 1)



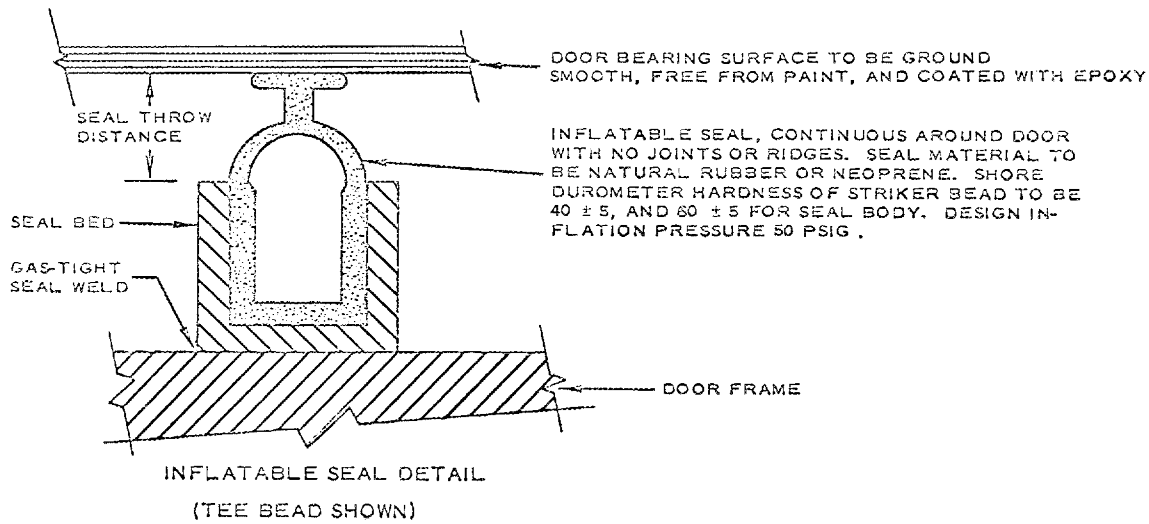




Fig. 9.7. Sliding Metal Door with Inflatable Seals. (From ref. 2)

require less machining of the sealing surfaces, since surface irregularities are more easily compensated for by the flexing caused by the internal seal pressure. On the other hand, commercially available inflatable seals require a complicated retaining groove that adds to the machining cost. Inflation of the seal is an additional step to be sequenced into the operation of the air lock, which complicates the air-lock control system or increases the probability that the door will be improperly sealed. The inflatable seal also has the disadvantages of being less durable, being limited to containment pressures below the inflation pressure, and being dependent upon a supply of air pressure to maintain the seal over extended periods of time.

### 9.2.3.3 Power Systems for Air-Lock Operation

Power operation may be employed for the following functions:

1. Actuating equalizing valves,
2. Actuating wedges or dogs on door,
3. Swinging large doors,
4. Raising and lowering bridge systems on either side of equipment locks.

The first three functions must be sequenced for proper operation, and, if automatic, can be accomplished by mechanical means, such as by the use of central gearing. This may be seen in Fig. 9.4. Mechanical gearing allows manual operation in case of power failure. The central mover can be powered either by an electric motor or by a motor-driven hydraulic pump. Power is usually provided by a combination of mechanical, electrical, and hydraulic equipment. In addition to sequencing requirements, it is necessary to be able to operate either door from inside or outside the containment vessel or from inside the lock. This requires three control panels and routing of control cables and tubing into and out of the lock. To minimize gearing and penetrations, the prime mover may be located inside the lock if space permits.

### 9.2.3.4 Interlocks and Interlock Bypasses

Interlocking devices to prevent both doors of a lock being open at the same time must be mechanical in nature to ensure their operation in the event of a power failure. Cams, gears, stops, and flexible shafts have been used separately and in combination to provide an interlock. Interlocks are provided with overrides or bypasses that will permit both doors to be open at the same time. These bypasses may be effected by access to a gearbox with a key or by use of special tools. Bypassing the interlocks, which invalidates the containment, must be done only during those periods when containment integrity is not required.

### 9.2.3.5 Equalizing Valves

Equalizing valves are required to equalize any differential pressure across an air-lock door so that it can be opened. Equalizing valves may be seen in Figs. 9.2, 9.3, and 9.4. These valves usually are quick acting to minimize the time required to open the doors. Plug valves, ball valves, and rapid-acting globe valves have been used. Care must be taken to assure the leaktightness, operability, and proper sequencing of these valves, since they are an integral part of the containment envelope boundary. The criteria for sizing the valve and pipe must include the length of time available for equalizing the pressure under both normal and accident conditions. This, in turn, is a function of the manning movements required, manpower costs, and the injury that might occur as a result of radiation dose, temperature, and rapid depressurization. If the lock is used for personnel access during pressure testing, it also may be necessary to establish a minimum pressure equalization rate to prevent injury due to rapid depressurization. Reference 3 allows depressurization from 14.3 psig to atmospheric conditions in 30 sec and recommends use of the procedures of ref. 4 for depressurization from higher pressures.

#### 9.2.3.6 Removable Floor Systems

A portion of the floor system or track used to transport equipment through an equipment lock must be removable to allow the door to open and close. This requires a short bridge on either side of the lock that may be power operated or removed by overhead hoist. These removable bridges and lock floors are usually designed for heavy loads, since a spent fuel cask and transfer car may weigh as much as 80 tons. These heavy loads require considerable reinforcing of both the lock and the vessel wall to limit the deflections and stresses caused by the moving load. It is generally not feasible to use special supports to keep the loads from being taken by the shell, since any fixed supports will restrain the shell against thermal or pressure expansion. Temporary supports can be used if necessary.

### 9.2.4 Doors in Low-Leakage Buildings

Low-pressure containment structures (below 5 psig) and negative-pressure confinement have been used at several plants, such as for the refueling building at Humboldt Bay, the Hanford New Production Reactor (NPR), and the Canadian Nuclear Power Demonstration plant (NPD). The following types of doors have been used with success.

#### 9.2.4.1 Quick-Acting Watertight Ship's Door

Doors of the type used on ships are used on the Humboldt Bay refueling building. Two such doors are used in series with a flexible cable interlock. Soft rubber gaskets are used for the door sealing surfaces. The door frame is mounted in steel plate that is seal welded to channels embedded in the 12-in.-thick concrete walls of the building. The building is maintained at a negative differential pressure of 0.25 in. $H_2O$.

#### 9.2.4.2 Bulkhead Door With Inflatable Seals

Bulkhead doors with inflatable seals are used on the NPR containment structure. The doors are provided with double perimeter pneumatic seals to permit checking for leakage through the seals by pressurizing the space between the two seals.

#### 9.2.4.3 Shielding Door

The Humboldt Bay refueling building contains two large double-leaf railroad access doors that allow a railroad car carrying a spent fuel shipping cask to enter the building. These doors are shown in Fig. 9.8. They are made of reinforced concrete 12 in. thick, and each weighs 10 tons. They are provided with compressible cellular neoprene gaskets that can be adjusted to improve the seal if necessary. The doors are operated by pneumatic cylinders to assist in moving the heavy weight and to provide proper compression of the seals.

NPR building 105-N has shielding doors with double pneumatic seals. These doors are not operated during reactor operation.

There are several types of low-pressure, low-leakage doors commercially available with which there has been sufficient experience to indicate that trouble-free operation can be expected. These doors are available with almost any combination of compressible and inflatable seals and in standard package units of door, frame, and locking hardware. They are best suited to very near atmospheric pressure conditions and to manual operation.

Koontz has reported data on four types of access doors and various other building components that were tested at relatively low differential pressures for leakage rate.[2] These data provide a basis for selection of the type and number of doors to fit the design allowable leakage rate for low-pressure containment and confinement structures.

### 9.2.5 Testing of Access Penetrations

#### 9.2.5.1 Shop Testing

Because of the tolerances and complexity of fabrication, most commercially available air locks and special doors are fabricated, postweld heat treated, and pressure tested with their mating surfaces of door frames in the shop. They are then delivered in working order, complete with ASME Code symbol, unless dismantling is required for shipping purposes.

#### 9.2.5.2 Field Testing

Both field-fabricated doors and shop-fabricated doors and locks are leak tested after installation by one or more of the local leak-testing methods described in Chapter 10. The air locks and doors are usually installed in the vessel before the pressure test and the initial integrated leakage rate test of the entire vessel are conducted, and thus they are fully tested (see Chap. 10).

#### 9.2.5.3 Retesting

Access penetrations, which by their nature preclude permanent closure by positive methods such as welding, are more subject to failure than static portions of the containment system. Therefore they require retesting to provide continued assurance of containment integrity. It is common practice to routinely inspect the seals of all access penetrations and retest them frequently. Large doors used for access only

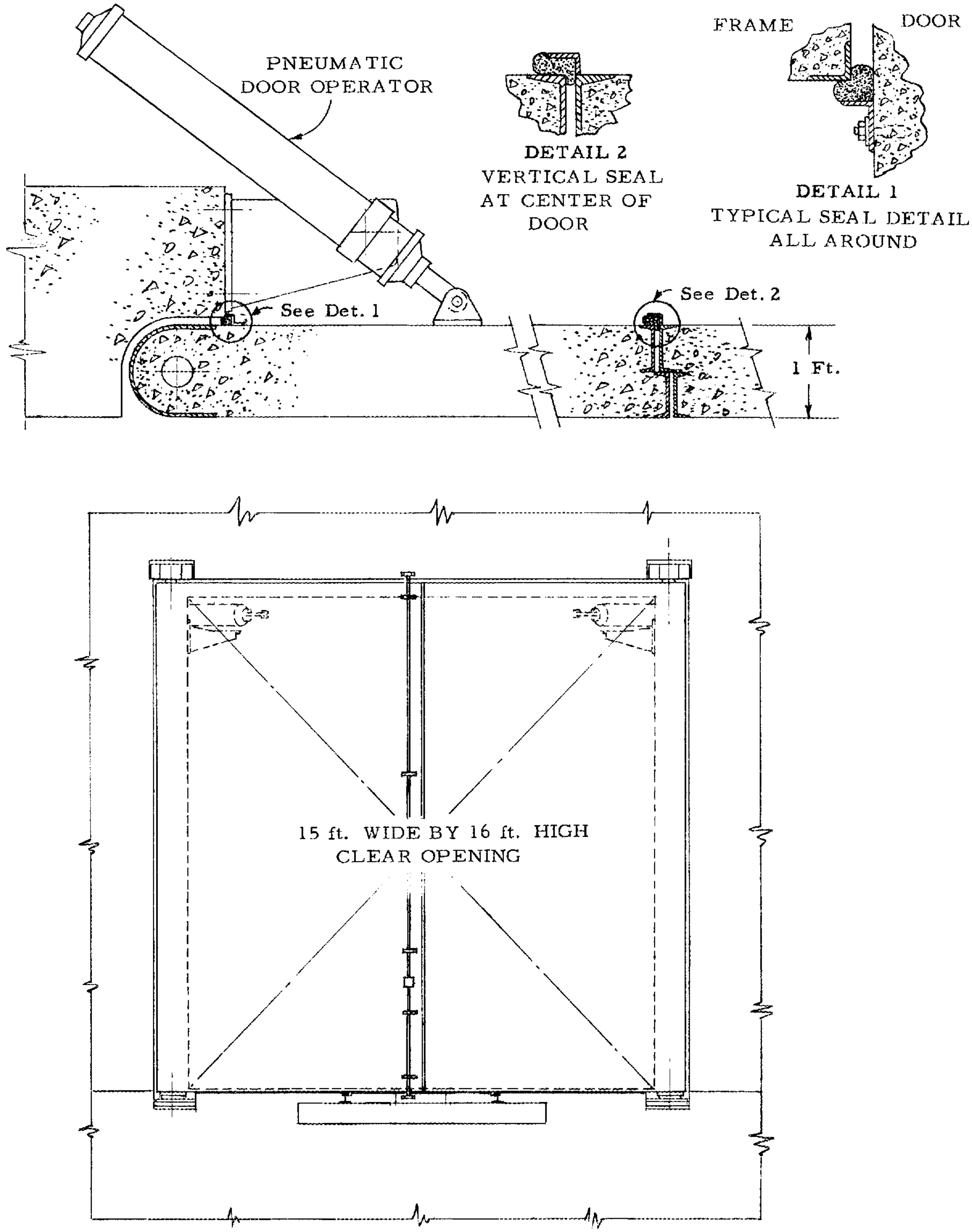



Fig. 9.8. Double-Leaf Railroad Door at Humboldt Bay Refueling Building.

during plant shutdown are usually leak tested each time they have been used and resealed.

## 9.3 PIPING PENETRATIONS

### 9.3.1 General

Penetration of the containment wall by piping of various types presents two related but quite different problems: (1) that of making a leaktight seal between the pipe and the containment wall, with provision for relative movement between the pipe and the wall if necessary, and (2) that of sealing the interior of the pipe by valves or other means to prevent leakage of radioactive material through the pipe to the environment in the event of an accident. The first of these problems is relatively straightforward and can be solved by proper mechanical design of the penetration assembly. The technical requirements are quite clear, and the basic problem is one of finding the design that can best meet these requirements at the least expense. Although some designs may become quite complex, many satisfactory designs have been developed and used with success in several different applications.

The second problem is that of determining the requirements that must be characteristic of the valve for sealing, such as the degree of sealing required, the method of sealing, the times at which the seals should be in effect, the provisions for leak testing, and the length of time permitted to make the seals if they are not in effect at the time of an accident. Pipes communicating directly between the containment interior and the outside atmosphere during reactor operation, as in the case of an open ventilation system in a conventional containment vessel, clearly need at least one isolation valve that can close quickly in the event of an accident. However, many pipes penetrating the containment wall terminate at either or both ends at closed, leaktight equipment or systems. If the systems and the connecting piping are designed for the containment design pressure or greater, they can be considered extensions of the containment system, and, conceivably, sealing of the piping might not be required. Between these extremes are many other conditions, each of which must be considered separately. In practice, isolation valves are usually used to minimize the extension of the containment boundary beyond the containment vessel wall, and all pipes penetrating the wall are grouped into just a few categories, each with different general requirements for isolation valves.

This section discusses the requirements for various types of piping penetrations and presents illustrations of some specific penetration designs. It also outlines some of the considerations involved and the practices commonly followed in specifying requirements for isolation valves.

### 9.3.2 Piping Penetration Designs

The basic requirements for piping penetrations, in addition to the normal piping criteria, may be listed as follows:

1. Maintain containment integrity,
2. Prevent excessive pipe loads from being transferred to containment system,
3. Prevent pipe from restraining the containment system during thermal or pressure expansions,
4. Minimize heat transfer into shell.

Maintaining containment integrity at piping penetrations requires that the piping itself be of high integrity and that an effective seal be formed between the pipe and the containment envelope. This is nearly always accomplished by welding steel piping to the containment vessel wall or to the liner in the case of a steel-lined concrete vessel. The welded attachment may be made either directly to a penetration nozzle or to an expansion joint that is welded to the nozzle. Penetration nozzles equipped with test caps are usually installed in the vessel by the vessel fabricator. Reinforcement of openings formed by nozzles is required in accordance with ASME Code Case 1272N-5 and Section III of the ASME Boiler and Pressure Vessel Code. Both the Code Case and Section III require all doors, nozzles, and opening frames to be preassembled into the shell plates and postweld heat treated as complete subassemblies for welding into the shell.

This simplest type of piping penetration is shown in Fig. 9.9, in which the pipe is butt welded directly to both ends of the penetration nozzle. However, this design requires very close tolerances on pipe spools, so an oversized nozzle and cap, as shown in Fig. 9.10 is often

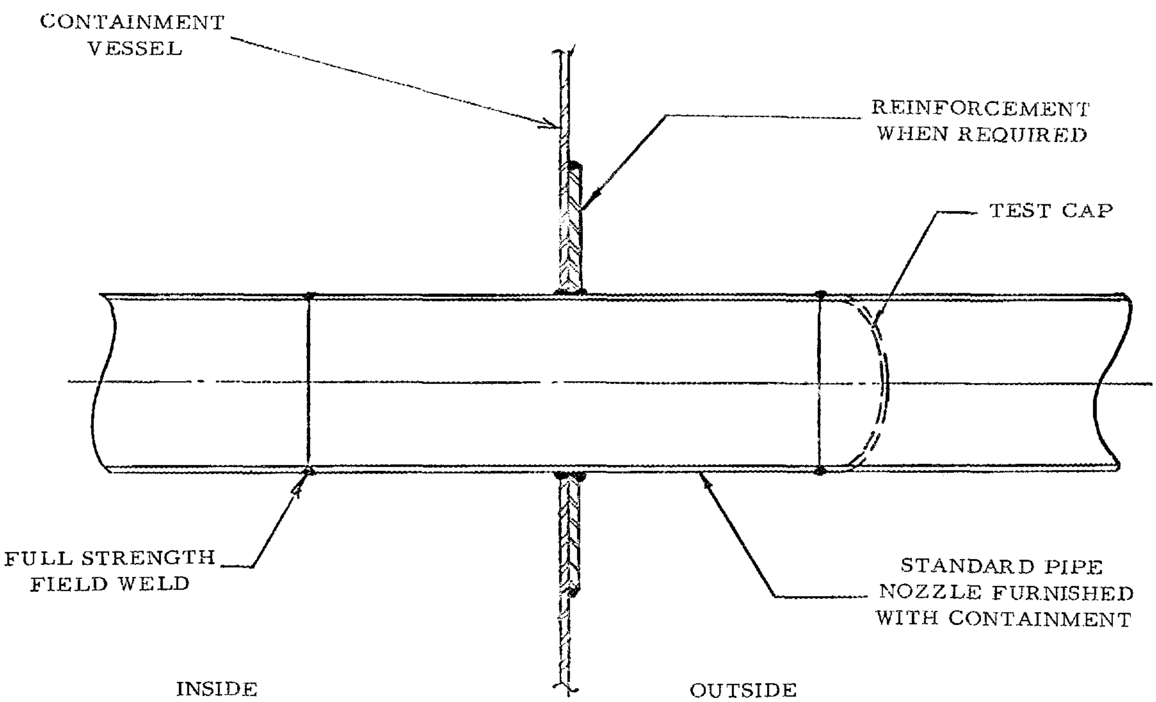



Fig. 9.9. Simple Piping Penetration.

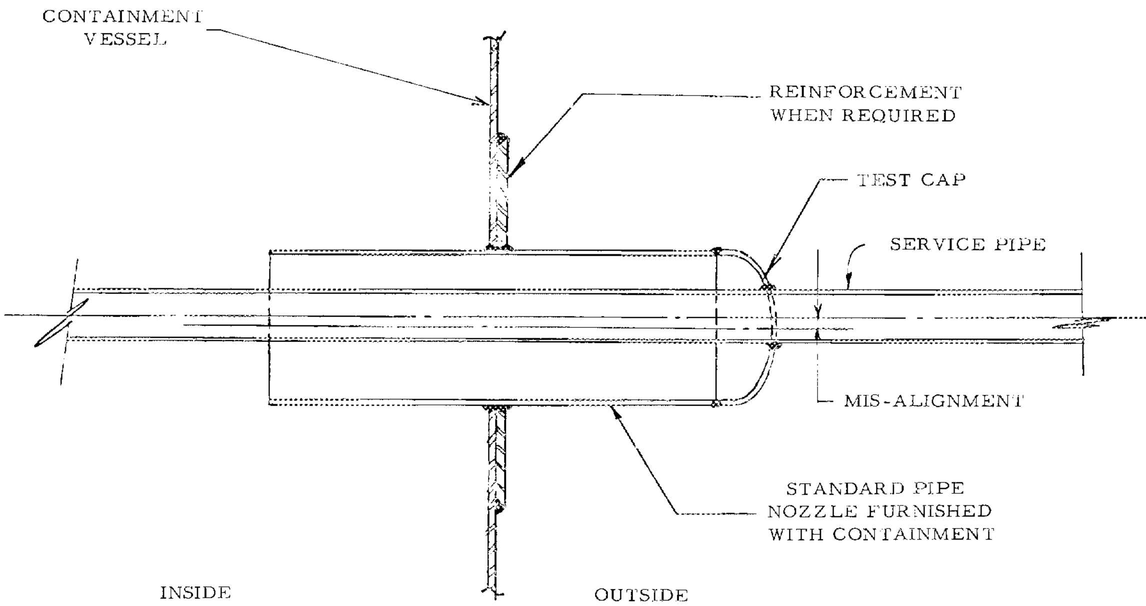



Fig. 9.10. Oversize Nozzle and Cap for Simple Piping Penetration.

used instead. This arrangement substantially eases the lateral tolerance requirements and essentially removes axial tolerance requirements, since the cap may be bored or flame cut to suit the field conditions.

Both of the foregoing types of penetrations act as anchor points for the piping systems. Therefore the reactions imposed on the containment vessel by the thermal expansion or contraction of the pipe should be carefully evaluated. Similarly, the effect of containment expansion or contraction on the piping system should be analyzed. When the reactions on the containment wall are too high or when the piping system cannot be anchored at the penetration, an arrangement permitting relative motion between the piping and the containment wall must be used. One common solution is the use of a metallic bellows expansion joint, such as that shown in Fig. 9.11, which is welded to the containment nozzle on one end and to a reducer and the penetrating pipe on the other. This design was used at the Dresden and Big Rock Point plants and for the 10-in. main steam lines of the NS Savannah. These expansion joints are usually of a standard type that is available through numerous suppliers, and they are fabricated in accordance with applicable codes. As a protection against physical damage, an external guard or shroud is recommended. Since metallic bellows joints will not tolerate torsional strain, the penetrating pipe must be restrained in such a manner that it cannot rotate about its centerline.

For high-temperature applications, the expansion and insulation requirements may be severe enough to require an elaborate solution that provides insulation as well as mechanical flexibility. An example of the solution used for the EBR-II is illustrated in Fig. 9.12.

If a main steam pipe failed where it passed through a bellows type of penetration, the resulting high instantaneous pressure could cause the bellows to fail. Such a failure could be particularly serious in a

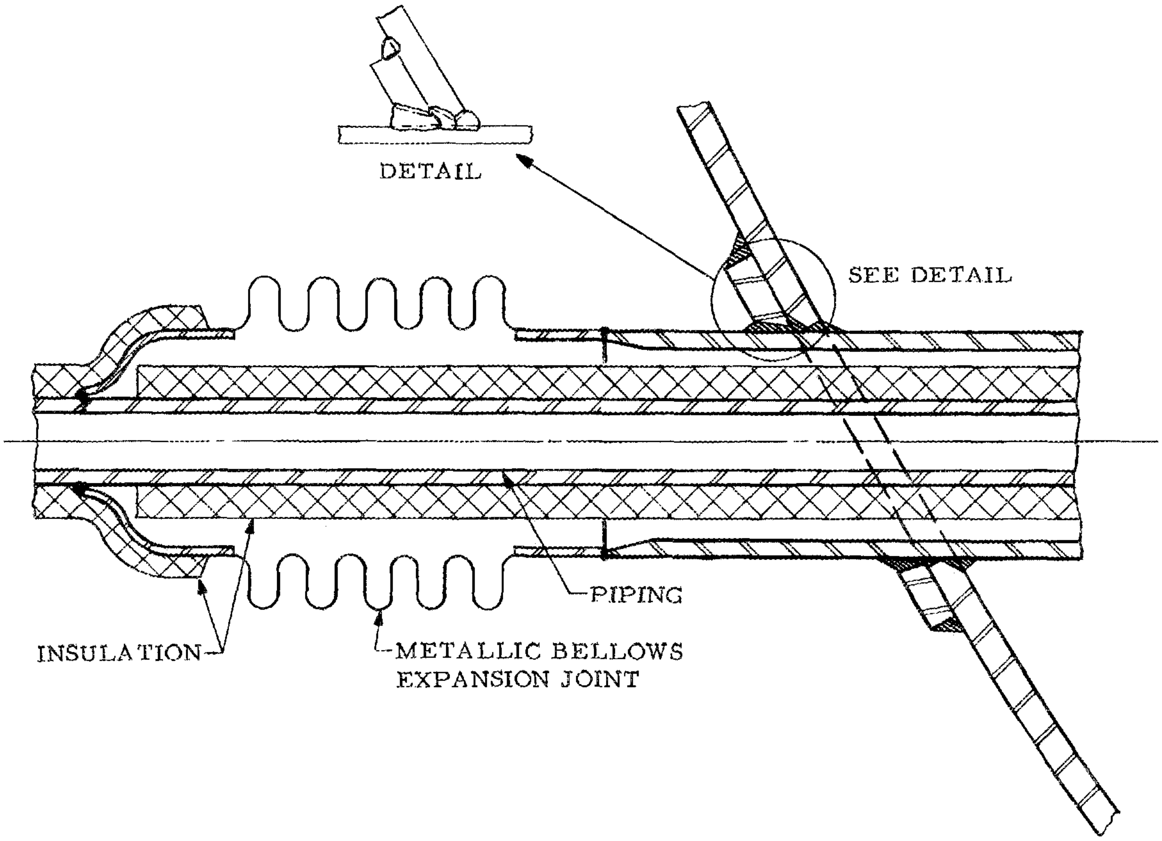
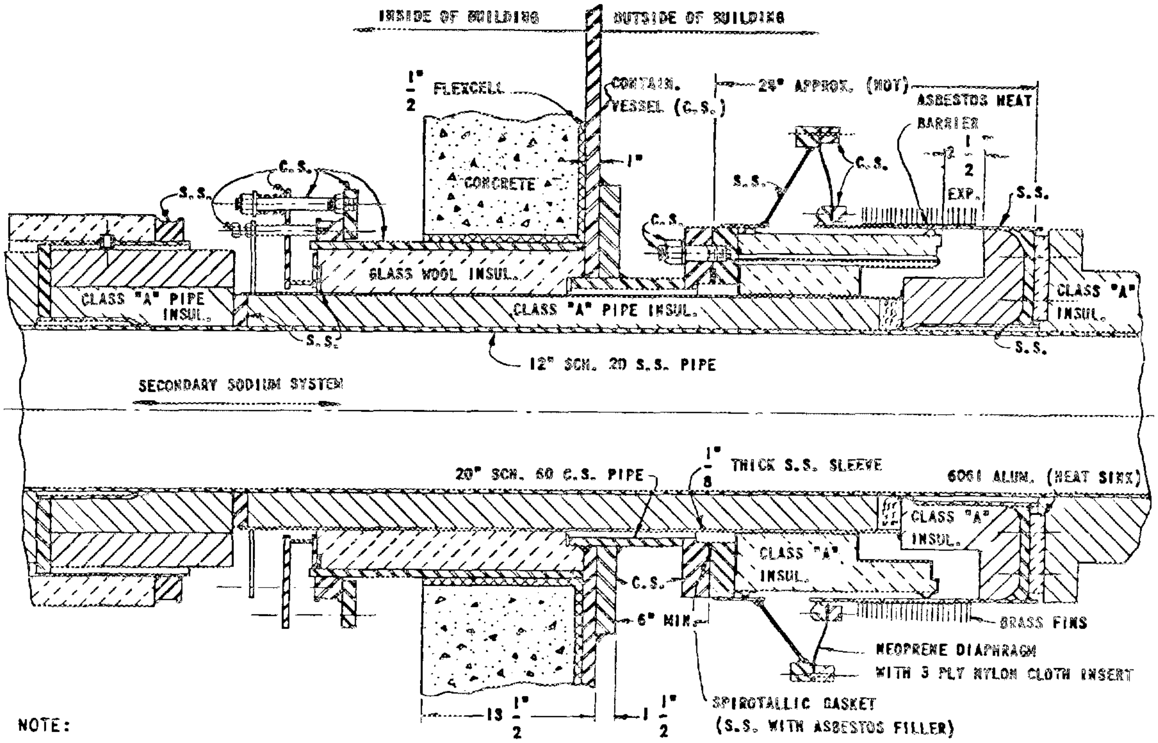



Fig. 9.11. Typical Pipe Penetration with Metallic Bellows. (From Chicago Bridge & Iron Company)

direct-cycle water-cooled reactor plant, since radioactivity would be released from the primary system at the same time that the containment integrity was breached. Moreover, subsequent closure of the steam isolation valve, if it were outside the containment boundary, would not stop the flow of primary steam out of the failed bellows. To use a bellows of low-pressure design on the Humboldt Bay plant, a guard pipe (designed for 1250 psig) enclosing the steam pipe was welded to the isolation valve body. The space within the guard pipe is vented to the containment system (the drywell in this case). Thus, the bellows would be protected from high-pressure steam until after it had been vented to the drywell. Figure 9.13 shows the detail of this type of penetration.

An additional consideration that must be examined with regard to boiling-water reactors or other direct-cycle types is the effectiveness and reliability of singly containment-connected steam isolation valves in retention of radioactivity in the event of an accident. It is to be noted that the singly containment-connected steam isolation valves having one barrier at the point of connection (e.g., the valve body) correspond to the two barriers, the primary coolant system and the containment for indirect-cycle reactors. Hence a scheme such as that depicted by Fig. 9.13 for boiling-water reactors should be shown to have comparable effectiveness to the conventional arrangement. It may be that consequences




NOTE:

1. THIS PIPE CARRIES SECONDARY SYSTEM SODIUM BETWEEN MAIN HEAT EXCHANGER AND STEAM GENERATOR. THE FLEXIBLE SEAL ARRANGEMENT SHOWN IS NECESSARY TO ACCOMMODATE PIPE THERMAL EXPANSION (IT BEING IMPRACTICAL TO PROVIDE AN EXPANSION LOOP INSIDE THE REACTOR BUILDING).
2. OPERATING TEMPERATURE RANGE OF THE Na, 500°F - 900°F

Fig. 9.12. EBR-II Sodium Piping Penetration. (From ref. 1)

of failure of singly containment-connected isolation valves are of such magnitude for some direct-cycle reactors that a design incorporating containment of the inner of the two steam isolation valves may be required.

Most components of piping penetrations are installed in the field to accommodate the fit-up of the pipe. After removal of the pipe cap or plate from the containment vessel nozzle, the nozzle is beveled, as required for welding, and the pipe and bellows are installed, fitted, and welded. Careful attention must be given to the weldability requirements of the transition welds between the different materials used in the penetration, bellows, and pipe. The field welds at both ends of the bellows are usually made under the requirements of the Code or Code Case applicable to the containment vessel, even though the Code jurisdiction terminates at the first circumferential joint outside the vessel. The weld between the pressure pipe and reducer must meet the welding standards of the ASA piping code.

When the piping penetrations also extend through adjacent rigid concrete structures, provisions must be made for relative motion between the piping and the concrete. Restraint of the nozzles or piping should be prevented by wrapping the embedded portion with a compressible material, such as Fiberglas or a similar material.

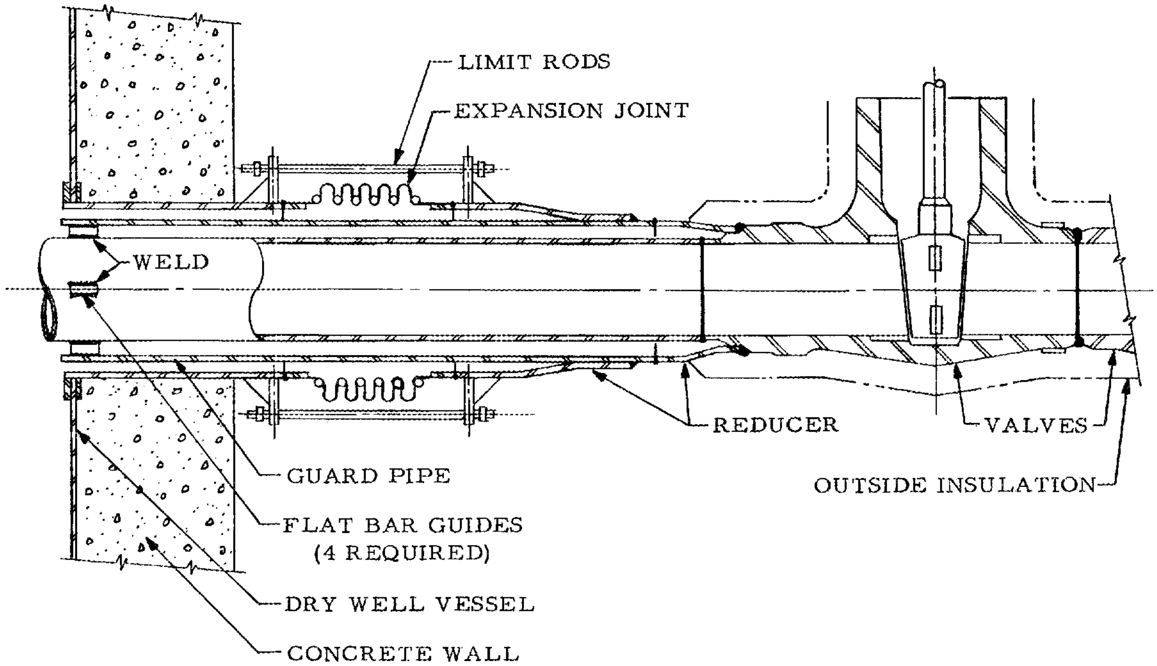



Fig. 9.13. Humboldt Bay Primary Steam Piping Penetration.

## 9.3.3 Isolation Valves

### 9.3.3.1 Criteria

The openings in containment structures caused by piping penetrations must be controlled, just as access penetrations are controlled, in order to avoid violation of containment integrity. Although some of these openings are normally closed, many must remain open or be opened occasionally if the reactor is to operate. However, since they are usually substantially smaller than access penetrations and transmit only fluids rather than solids, they can be closed much more easily and rapidly when required. These openings are often separated from both the primary coolant system and the containment atmosphere by at least one solid metallic barrier, such as a heat exchanger shell or tube wall, which also must fail if radioactive material is to be released through the piping penetration. Because of these factors, normally open or occasionally opened piping penetrations are allowed provided they are equipped with appropriate isolation valves. The number and types of isolation valves used and the closure speeds required depend upon the amount and type of radioactive material potentially available to the fluid being transmitted, the time dependency of this source entering the fluid, the transport characteristics of the fluid, the degree of continement of the fluid and its contained radioactive material in any secondary confinement systems, and the consequences of failure of or leakage through an isolation valve under accident conditions.

Because of the different radioactive material source and fluid transfer characteristics associated with each type of reactor and plant design, the isolation valve requirements must be evaluated for each specific application. However, although exceptions may exist, the following criteria represent the practices usually followed with respect to the number and location of isolation valves to be used in piping that penetrates the containment boundary.

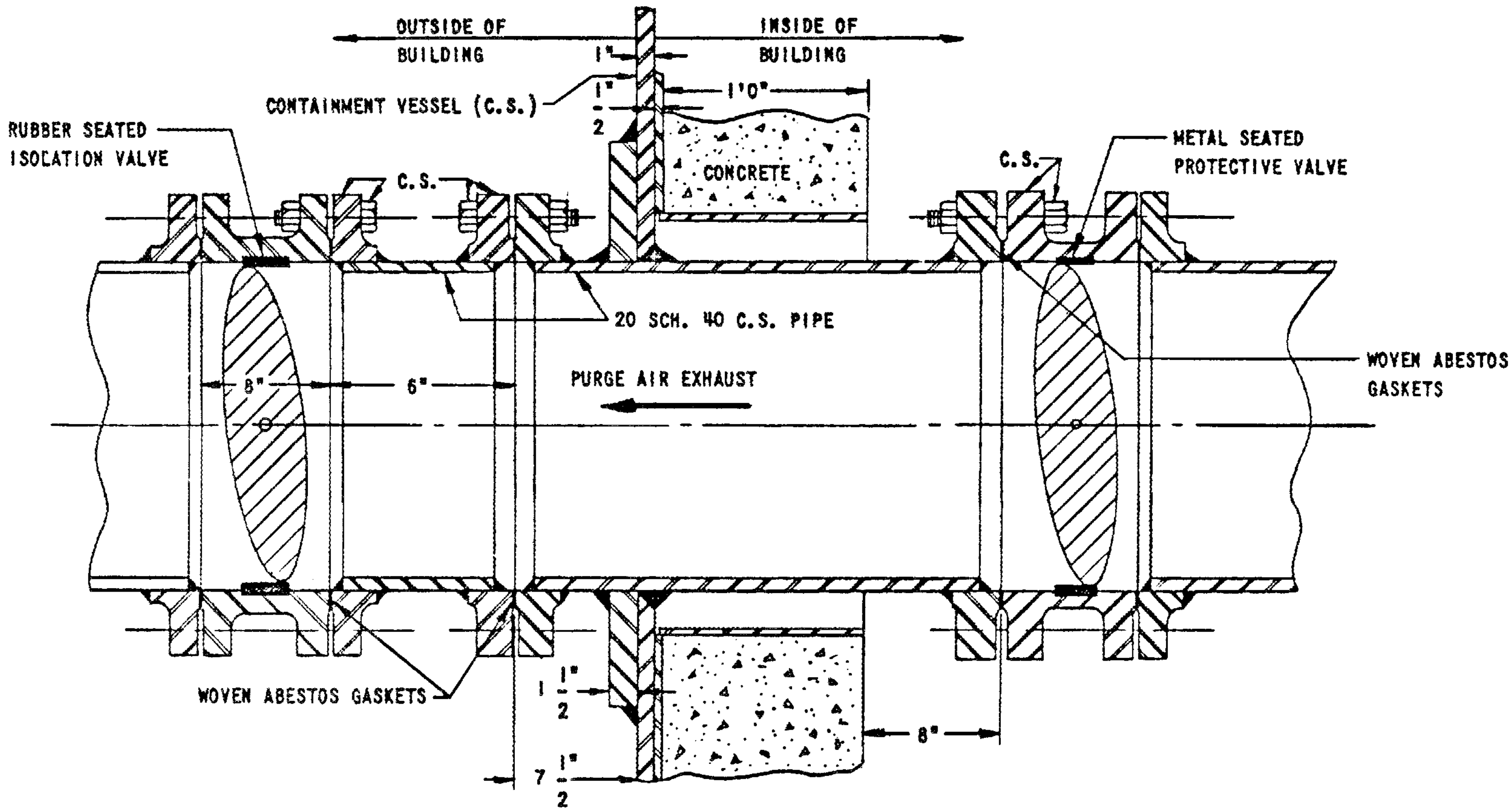
1. Normally Closed Lines. Lines that are normally closed need only a single isolation valve. A lock or seal or interlock, if remotely actuated, should be provided to prevent this type of valve from being opened during reactor operation or during otherwise potentially hazardous situations. Even though normally closed, lines routinely containing very highly radioactive fluids are often equipped with multiple valves to guard against accidental opening and to provide greater assurance of leaktightness.

2. Normally Open or Occasionally Opened Lines.

Lines that connect to the primary coolant system are normally provided with two isolation valves. For incoming lines (makeup, feedwater, energency cooling, control rod cooling), one or both of these may be check valves. The valves should be located so that one is inside and one outside the containment. At least one should close automatically to prevent flow reversal. For outgoing lines (main steam lines in direct-cycle plants, purification system, emergency cooling), one valve is also normally placed on each side of the containment wall. At least one of these valves should close automatically upon receipt of a signal indicating a system failure. On the Humboldt Bay plant, two tandem isolation valves are located on the main steam line just outside the drywell, but, as shown in Fig. 9.13, a guard pipe extends the drywell barrier to the first valve and thereby, in effect, makes one valve body part of the wall and one valve external to the wall.

Lines that are open to the containment system (ventilation and purging systems, containment spray) are normally provided with two valves in series that should be placed on each side of the containment wall, as shown in Fig. 9.14. At least one should close automatically upon indication of a system failure. Ventilation system valves, which may be somewhat less positive in closing because of their greater dimensions, often are both automatically closed at the same time.

For lines that connect to closed loop systems within the containment system, no generalizations are possible. Since, by definition, these penetrations are separated from the containment atmosphere and the primary system by a continuous barrier, such as a pipe wall, heat exchanger tubing or casing, pump wall, etc., the need for the further protection provided by an isolation valve is dependent upon the vulnerability of the interior barrier to failure, the direction of flow likely with failure, and the radioactive material transport likely if a failure occurs. For example, the Yankee PWR and the NS Savannah have automatic isolation valves on all containment piping penetrations used during operation.[5] In contrast, the Peach Bottom helium-cooled reactor, where the containment pressure (only 8 psig under the worst accident conditions) is less than the pressure in most secondary systems, does not employ automatic isolation valving on certain service and cooling water systems that do not involve significant radioactive transport and which are not likely to fail simultaneously with the maximum credible accident.[6] However, automatic isolation valves




NOTE: 1. THIS PIPE EMPLOYED ONLY TO PURGE BUILDING ATMOSPHERE IN EVENT OF SMALL SODIUM FIRES OR OTHER MINOR AIR-CONTAMINATION ACCIDENTS NOT INVOLVING SIGNIFICANT RADIOACTIVITY.

2. BOTH VALVES NORMALLY CLOSED. IF OPEN TEMPORARILY FOR PURGING, BOTH CLOSE AUTOMATICALLY IN EVENT OF MAJOR INCIDENT.

3. METAL SEATED VALVE NOT GASTIGHT; ITS PURPOSE IS TO PREVENT SIGNIFICANT COMMUNICATION OF BUILDING ATMOSPHERE WITH RUBBER SEATED ISOLATION VALVE WHICH IS GASTIGHT.

Fig. 9.14. EBR-II Ventilation System Isolation Valves. (From ref. 1)

are provided on the Peach Bottom main steam secondary coolant system, since a substantial failure in the main heat exchanger is considered credible.

9.3.3.2 Types of Valves

The types of isolation valves selected depend upon the service function to be performed. Generally, the basic characteristics will be the same as for standard valving used in other commercial applications. However, specifications for isolation valves used in nuclear systems may vary considerably, depending upon their specific application. For conventional plants, the designer may need to specify only the fluid, pressure, temperature, flow rate, applicable code, type of driver and controller, and the type of connection to the piping system. In addition, nuclear isolation valve specifications may also include the permissible leakage across the valve seat and through the stem seal, the expected pH, conductivity, radioactivity level and type of the fluid, future maintenance and inspection procedures that the valve design must permit, cleaning precautions, alloys or alloying elements to be avoided, and the opening and closing times. Much greater reliability is required for isolation valves that affect nuclear hazards than is required for most conventional valves. Hence, isolation valves require, in addition to stringent specifications, careful scrutiny of the fabrication, inspection, and installation phases. The types of valves commonly used as isolation valves and the types of systems for which they are most often used are indicated in Table 9.1.

Requirements for surveillance and maintenance should not be overlooked in the selection and installation of valves. For example, the design should permit determination that butterfly and check valves seat properly, that the condition of the diaphragm of a diaphragm valve is satisfactory, and that blockage by dirt and crud is not likely.

Table 9.1 Types of Isolation Valves

| Valve Type | Service | | |
|---|---|---|---|
| | Water | Steam | Ventilation |
| Globe | X | X | |
| Gate | X | X | |
| Butterfly (metal seat) | X | | |
| Butterfly (rubber seat) | | | X |
| Rotating plug | X | X | |
| Ball | X | X | |
| Diaphragm | X | | |
| Check valve | X | X | X |

The performance parameters of primary importance in selecting containment system isolation valves are leakage rate and rate of closure. Leakage in valves includes both the leakage across the seat within the valve body and across the seal around the valve stem. Complete elimination of all leakage across the valve seat is virtually impossible, especially under conditions of frequent operation and high-temperature service. For this reason, isolation valves are not normally used for fluid flow control and are not normally called upon for frequent operation. In discussing the application of valves for water-cooled reactor service, Gruenwald[7] states that standard water valves of "commercial tightness" have a seat leakage rate of no more than 10 cubic centimeters per hour per inch of pipe diameter, while valves designed for nuclear service are available that leak no more than 2 cubic centimeters per hour per inch of pipe diameter.

Stem leakage can be reduced to zero by using bellows around the stem or by using diaphragm valves, but these are expensive and unreliable in large valve sizes and for high-pressure systems. Consequently, limited stem leakage is usually accepted, and means are provided for collecting this leakage if it could be hazardous.

The valve closure rate obtainable is primarily a function of the drivers required to overcome the inertia of the moving valve parts and the hydraulic forces of the moving fluid. However, determining appropriate closure speeds requires evaluation of the system's capability to cope with valve closure, as well as evaluation of the closure rate necessary for the maintenance of containment integrity. Sudden closure of a valve might create a water hammer or even cause feedback effects to the nuclear system. Closure times on main steam lines may be as long as 40 sec (Big Rock Point), while ventilation system closures may be made in as little as 6 sec (Peach Bottom). Check valve operation is possible in substantially less than 1 sec if there is a strong and immediate tendency for flow reversal. The cost of an isolation valve increases considerably as the specified closure time is reduced. Therefore, a shorter closing time than is actually required should not be specified. The 10-in. steam stop valves just outside the containment on the NS Savannah were, for example, originally capable of closing in less than 1/3 of a second. This closure time was later determined to be not only unnecessary but also detrimental to seat maintenance.

### 9.3.4 Penetrations and Isolation Valving in Conventional Structures

When conventional metal panel and concrete structures are used for containment or confinement systems, it is necessary to use special provisions for piping entering and leaving the building. A penetration through a concrete wall is shown in Fig. 9.15, and a penetration in a metal panel is shown in Fig. 9.16. Figures 9.17 and 9.18 illustrate piping penetrations through two types of roofs. In addition to sealing the obvious pipes, it is also necessary to isolate floor drains by running traps of sufficient depth or by installing isolation valves.

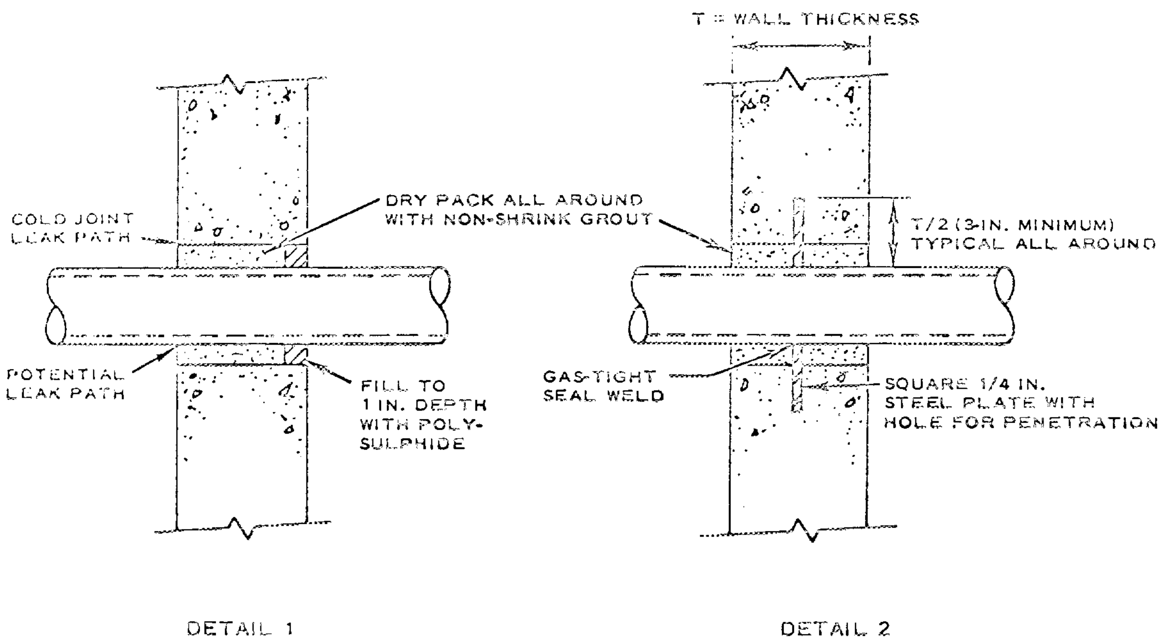



Fig. 9.15. Piping Penetrations Cast in Concrete Wall. (From ref. 2)

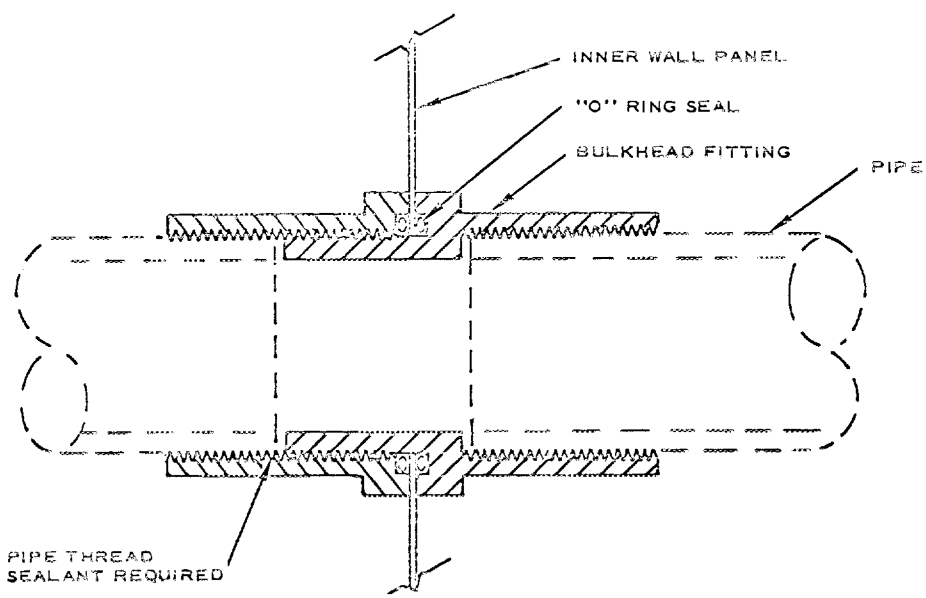



Fig. 9.16. Piping Penetration in Metal Panel Wall. (From ref. 2)

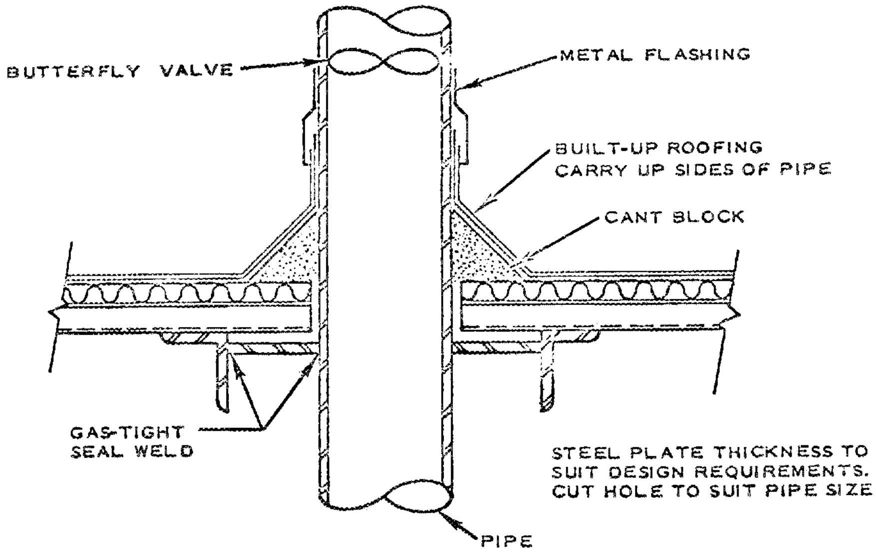



Fig. 9.17. Piping Penetration Through Metal Deck Roof. (From ref. 2)

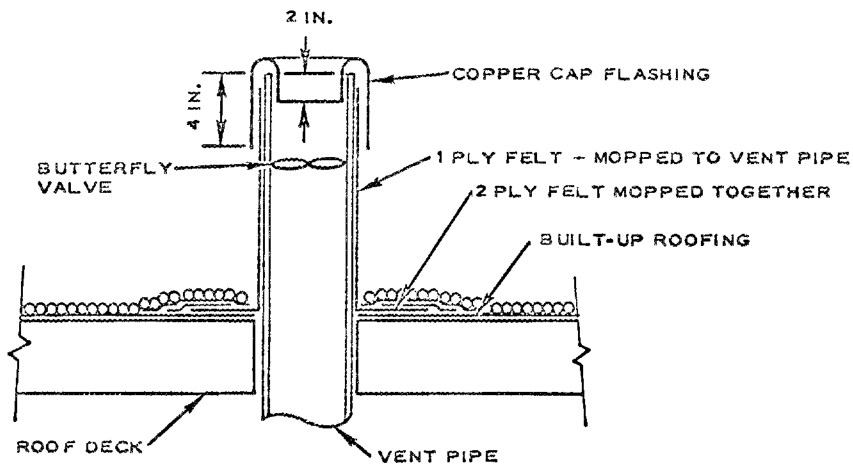



Fig. 9.18. Vent Pipe Penetration Through Builtup Roofing. (From ref. 2)

## 9.3.5 Testing

The previously listed piping criteria require that the penetration not leak and not transfer excessive pressure or thermal loads to the containment system. Leak testing of the penetration assembly should be performed in the manufacturing shop whenever possible but should always also be done after the assembly has been installed in the containment system in order to test the field welds. The penetration nozzles are normally provided by the vessel fabricator and are both pressure tested and leak tested along with the vessel as part of the vessel proof tests.

A second integrated leakage-rate test of the entire containment vessel should be performed after all penetration assemblies have been installed, and each penetration may be leak tested individually during or just prior to this test. Local leak testing is usually accomplished by pressurizing the containment system to some nominal pressure, generally around 5 psig, and applying a soap solution over all external joints of the penetration assembly. Leaks will be indicated by the formation of soap bubbles. Local retests should be performed occasionally of areas subject to deterioration; however, the welded joints of most piping penetrations would not be expected to develop leaks after once found to be leaktight. Leakage testing is discussed in more detail in Chapter 10.

Tests demonstrating that excessive pressure or thermal loads are not transmitted to the containment system are normally not possible because of the lack of ability to simulate all the thermal, hydraulic, and pneumatic conditions that might ultimately exist. Inspections should be performed to assure that the specified tolerances have been provided and that appropriate insulation has been properly installed.

Isolation valves are frequently tested prior to their installation because of the difficulty involved in obtaining subsequent absolute leakage-rate measurements. However, when possible, adulterant gas, pressure decay, or pressure buildup leakage tests are conducted upon completion of construction to reverify the operability and degree of leaktightness of the valve in its installed condition. Soap bubble tests are of value in determining valve stem leakage but are of little value in valve seating tests, even if the seat is reasonably accessible by breaking adjacent flanges.

## 9.4 FUEL TRANSFER PENETRATIONS

### 9.4.1 General

Some containment system designs include provisions for transferring spent fuel removed from the reactor during refueling operations through the containment wall to a spent fuel storage pool outside the containment. In some cases, the storage pool is connected by the fuel transfer tube to a refueling pool at a higher elevation within the containment vessel, and a lock arrangement is necessary to prevent water from flowing out of the higher pool into the lower one. Even when there is no elevated pool and no need for a water lock, an air lock still may be required to maintain containment integrity if the fuel is transferred during reactor operation of if a fuel meltdown accident during refueling operations could release substantial quantities of radioactive materials. Additional mechanisms may be required to change the direction of the fuel element as it is transferred or to provide cooling of the element during transfer. Heavy shielding of the entire transfer tube is also required. Fuel transfer mechanisms may therefore be quite complex devices with some features in common with both access penetrations and piping penetrations.

### 9.4.2 Fuel Transfer Mechanism Designs

Fuel transfer mechanisms are provided at the Dresden, Yankee, and EGCR plants. The Dresden design is shown in Fig. 9.19. The 42-in.-diam vertical fuel transfer tube is closed by means of a large gate valve located inside the containment vessel and above the water level normally maintained in the spent fuel storage pit during reactor operation. Fuel is lowered vertically through the transfer tube into the basket carrier and is transferred horizontally by the carrier into the storage pit. The carrier makes a seal with the bottom of the transfer tube and thus forms the second valve of a water-lock arrangement to maintain the proper water levels in the two pools.

The Yankee fuel transfer chute, shown in Fig. 9.20, has a gate valve at the lower end of the 12-in.-diam pipe, below the level of the water in the spent fuel storage pit, and approximately 50 ft outside the containment wall. A flapper valve, the second valve of the lock arrangement, is located at the top of the chute where it leaves the pool above the reactor. During reactor operation the chute is sealed by a solid plate inserted between gasketed flanges just outside the containment vessel wall.

The EGCR spent fuel transfer mechanism is shown in Fig. 9.21. This mechanism can transfer spent fuel either from the service machine above the reactor or from the charge machine below the reactor into the discharge chute and then into the spent fuel storage basin. The transfer operation is normally carried out in air, but provision is made for a water spray during emergencies. The discharge chute contains double doors and serves as an air lock to provide continuous maintenance of containment integrity.

As illustrated by each of these examples, a double valve arrangement should always be provided to prevent violation of the containment in the event dirt or debris prevents single valve closure.

## 9.5 ELECTRICAL PENETRATIONS

### 9.5.1 General

Hundreds of electrical conductors of various types must pass through the containment wall in a typical nuclear power plant and each one must be sealed to prevent leakage. Although these are static seals and individually may not present especially difficult sealing problems, the large number of possible leakage paths and the care that must be taken to ensure that each one is adequately sealed make the electrical penetrations an area of major concern in all containment systems. In addition to meeting normal electrical requirements, electrical penetrations must be able to maintain containment integrity over long periods of time under normal operating conditions and over shorter periods under the much more severe conditions that would result if an accident occurred.

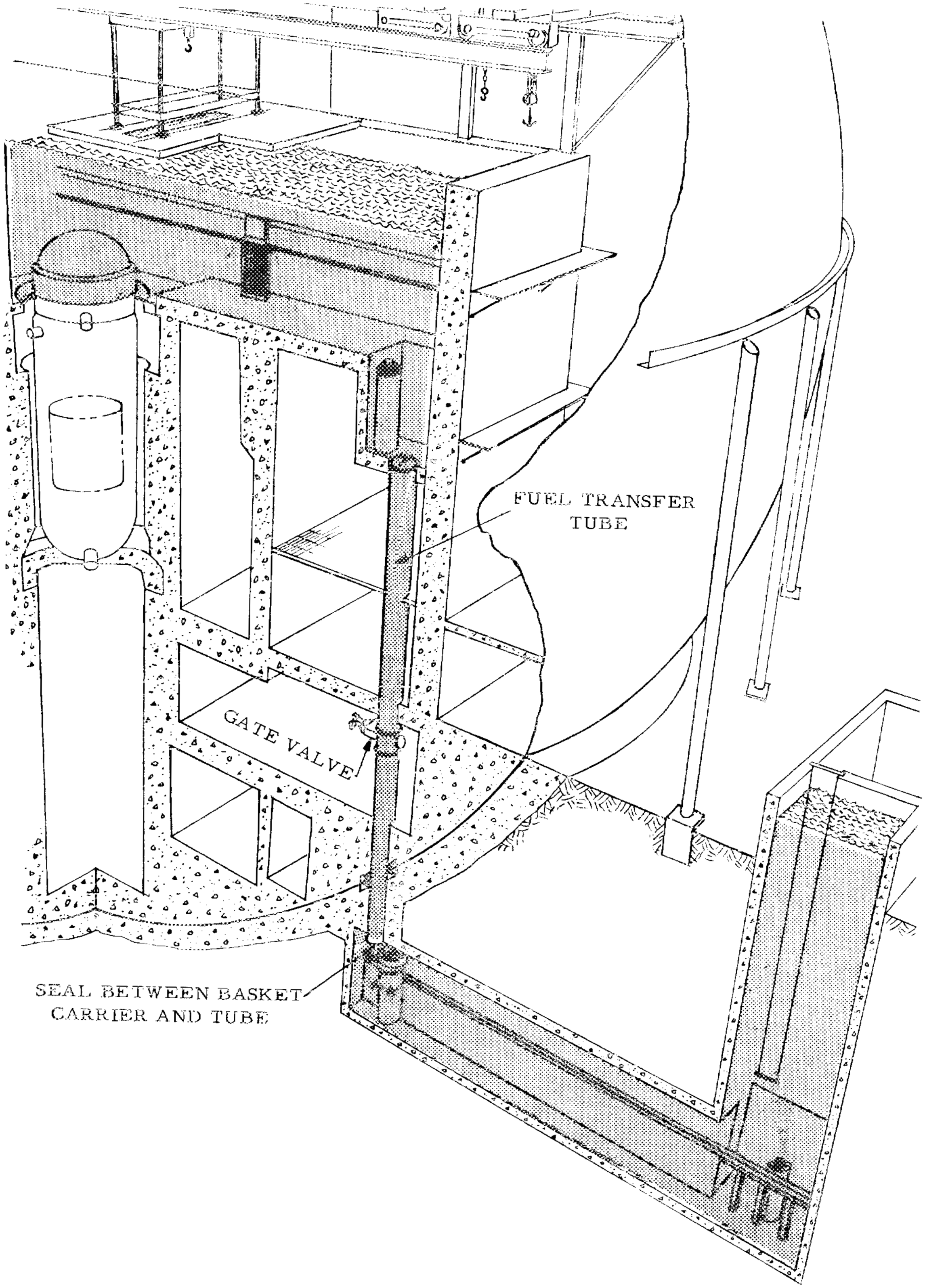



Fig. 9.19. Dresden Fuel Transfer Penetration.

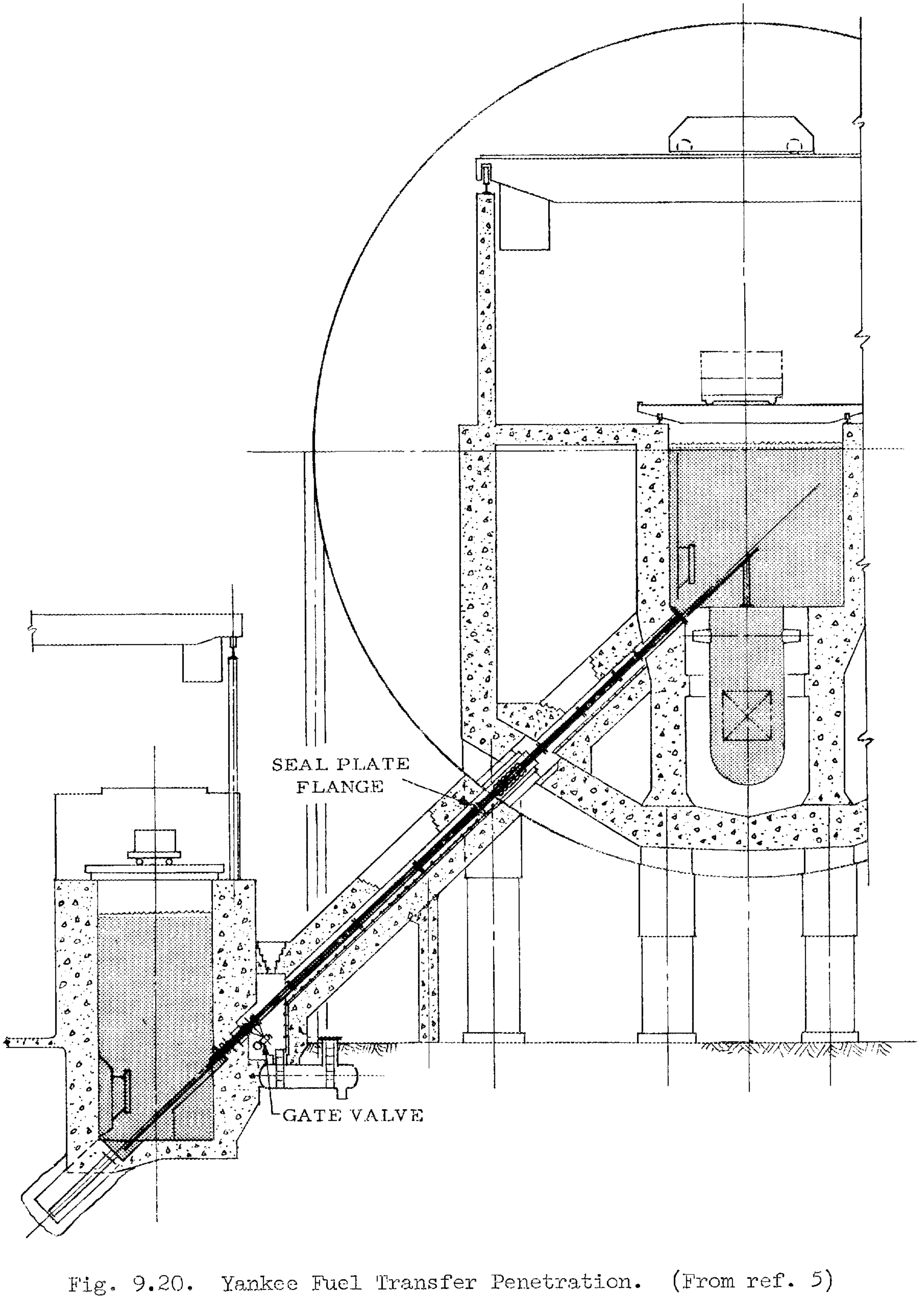



Fig. 9.20. Yankee Fuel Transfer Penetration. (From ref. 5)

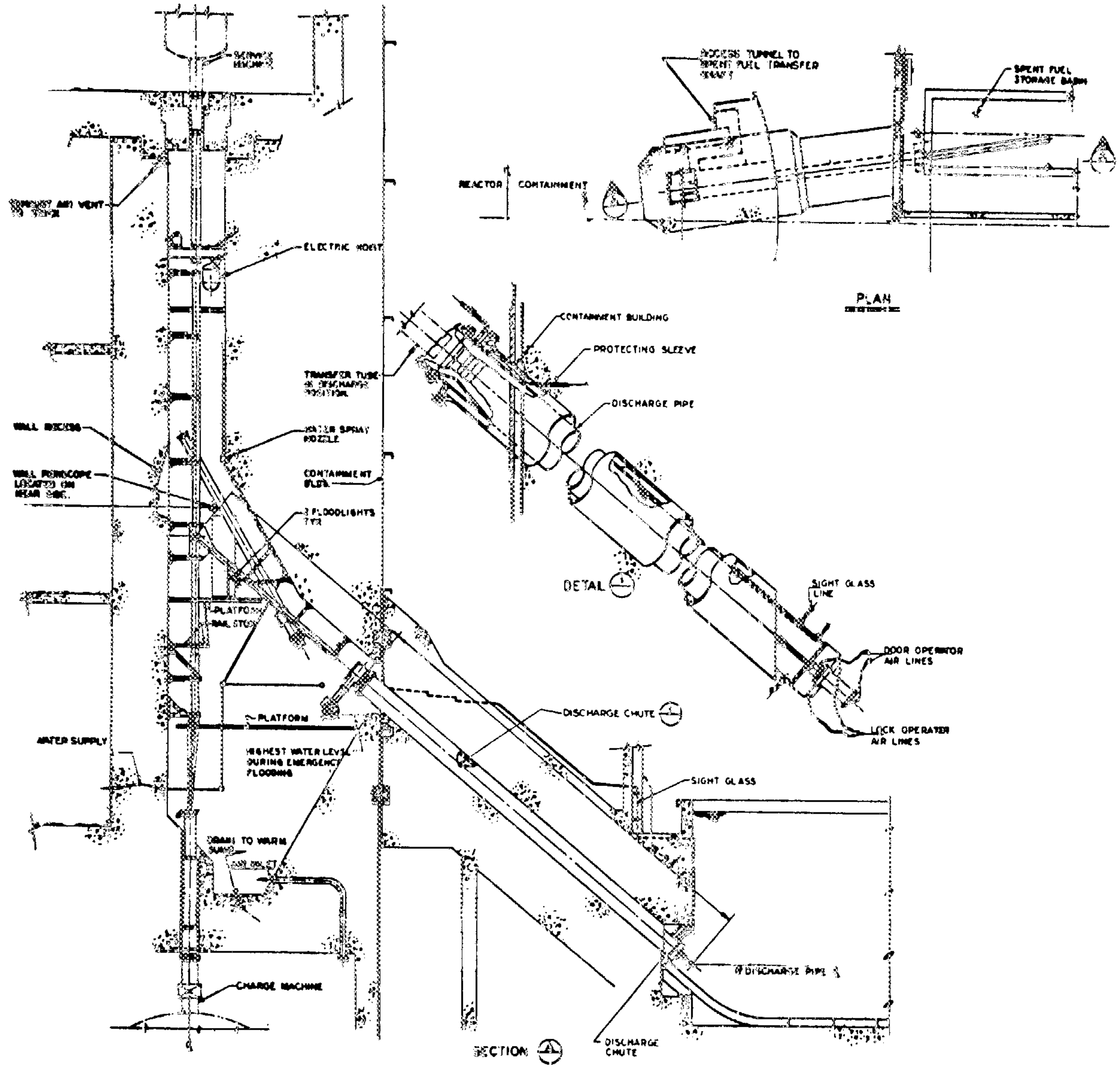



Fig. 9.21. EGCR Spent Fuel Transfer Mechanism.

Leakage through an electrical penetration could occur in any one or a combination of the following locations:

1. Between strands of multiple-strand conductors,
2. Between the conductor and its insulation,
3. Through the insulation or jacketing materials or between the insulation and any jacketing material,
4. Between the insulation or jacket and any potting, packing, or filler material,
5. Through the potting, packing, or filler material.
6. Between the filler material and the penetration mounting plate, pipe, or other surface.

The electrical cable used and the design of the penetration assembly must provide for sealing each of these possible leakage paths.

### 9.5.2 Electrical Cables

The first three of the possible leakage paths listed above are through the electrical cable itself. Therefore, even if an electrical penetration design provides for adequate sealing around the outside of the cable, leakage may still be possible unless a suitable type of cable is selected or additional provisions are taken to minimize these sources of leakage.

Solid conductors are frequently used to eliminate the possibility of leakage through the strands of multiple-strand conductors. Where stranded conductors or multiple conductor cables are used, leakage sometimes can be minimized by using cable in which the interstices have been filled with sealing compounds during manufacture. Cable of this type is available on special order for certain applications and for certain ranges of design temperature and pressure.

Leakage between the conductor and its insulation is minimized by using insulation materials that tightly bond with the metallic conductor. Thermosetting materials are preferred, since they offer greater resistance to flow under the elevated temperature and pressure conditions that could occur during an accident. Use of cable containing a sealing material may also reduce leakage between the conductor and the insulation.

Leakage through the insulation is minimized by using cable with a nonporous solid elastomeric insulation material. If cables are used that have woven fabrics or other porous insulation materials, such as magnesium oxide, special precautions should be taken to minimize leakage by impregnating the porous insulating materials with nonporous filler material or by replacing the porous material at the point of containment penetration. In order to assure that the provisions used to minimize leakage are adequate, it may be desirable to conduct leakage-rate tests on the cable prior to installation.

If conventional insulated cables are used in the penetration assembly, leakage between the strands of a multiple-strand conductor, between the conductor and insulation of stranded or solid conductors, and through the insulation may be minimized by applying a potted seal external to the penetration assembly. A seal of this type, which was used on the

Peach Bottom plant, is shown in Fig. 9.22. A penetration assembly used on the EBR-II, in which epoxy was used to seal the ends of the MgO insulation and the cable sheath on a solid conductor, is shown in Fig. 9.23.

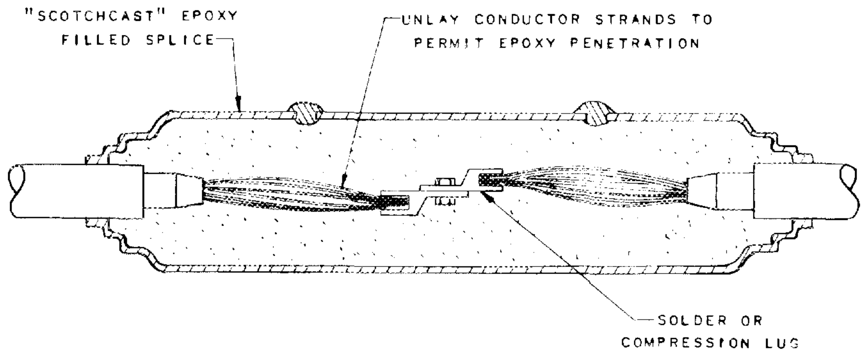



Fig. 9.22. Cable Splice and Gas Stop Used at Peach Bottom. ("Scotchcast" is manufactured by the Minnesota Mining and Manufacturing Co.) (From ref. 6)

### 9.5.3 Electrical Penetration Assemblies

Various electrical penetration designs have been used to meet a wide range of requirements in the containment vessels built to date. Selection of a particular type of penetration for a specific application depends on many factors, including the type and size of the conductor, the current and voltage to be carried, the number of penetrations required, the design temperature and pressure, and the cost of both materials and installation. Exceptional care in design and fabrication of high-pressure penetrations (above 5 psi) is necessary. For these applications, conventional cable penetrations should not be used.

The electrical property requirements of electrical penetrations vary substantially, depending upon the application. The types of cables and wires penetrating the containment wall include very low-power and noise-sensitive coaxial instrument leads, control wiring, cable supplying power for operation of equipment within the containment, and in some cases, relatively high-voltage power cables for transmitting large quantities of power generated within the containment. Transmitting power through the containment wall sometimes imposes an additional requirement, that of minimizing induction currents in the steel plate or the reinforcing steel of the containment structure.

Because of the large number of electrical conductors that normally must penetrate the containment wall, they are frequently grouped so that one penetration assembly may contain many individual conductors. The penetration assemblies often are mounted on plates or cylinders that are then attached to the nozzles in the containment vessel. This minimizes the number of separate penetrations in the containment wall.

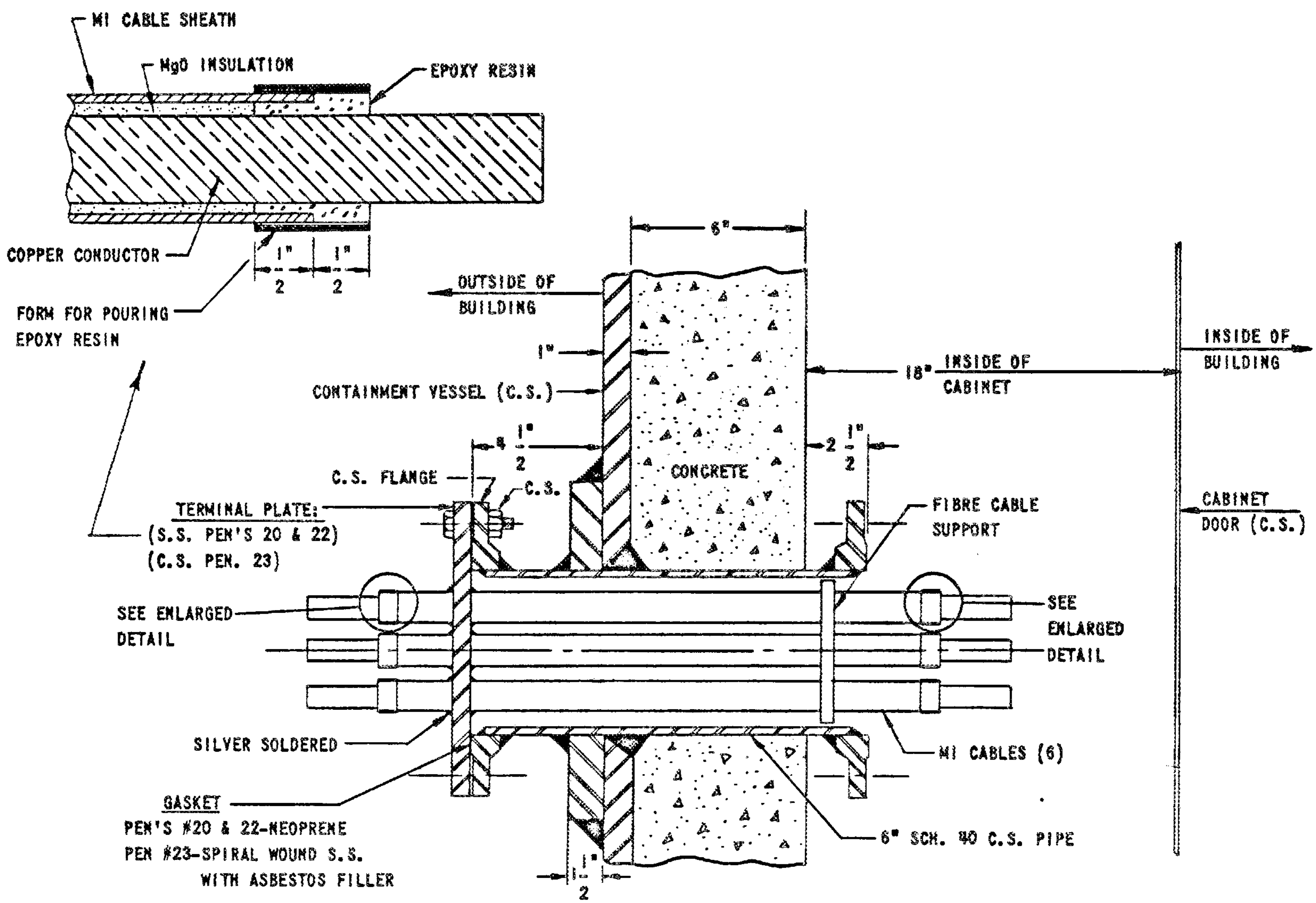



Fig. 9.23. EBR-II Electrical Penetration. (From ref. 1)

Penetration assemblies frequently are prefabricated in a shop and installed directly into nozzles in the containment vessel wall. The cables are then connected to both sides of the penetration. To permit continuous cable runs and thus minimize the time required to make cable connections, penetration designs have been used in some cases in which the seals are made in the field after all the cables have been pulled through the penetration opening. Penetrations of this type must use cable either sealed internally during fabrication or provided with a potted seal, as discussed in Section 9.5.2, above.

There are five general types of electrical penetration in use in one or more of the applications discussed above. Each of these types is discussed briefly and illustrated below. None of these types can be given specific ratings of voltage, current, temperature, etc. or related to a specific application because a variety of materials or compounds can be used with each of them to suit a wide range of requirements.

### 9.5.3.1 Packed

Penetration assemblies of the packed type are adaptations of the stuffing-gland principle. A resilient packing material is compressed against the outside of the cable to be sealed and against the inside of the surrounding tube. Continuous runs of cable can be used with this type of seal. Internal sealing or other means of preventing leakage through the cable must be used. A penetration assembly of this type, in which two packing glands are used, is shown in Fig. 9.24. This particular design was used on the Fermi plant.

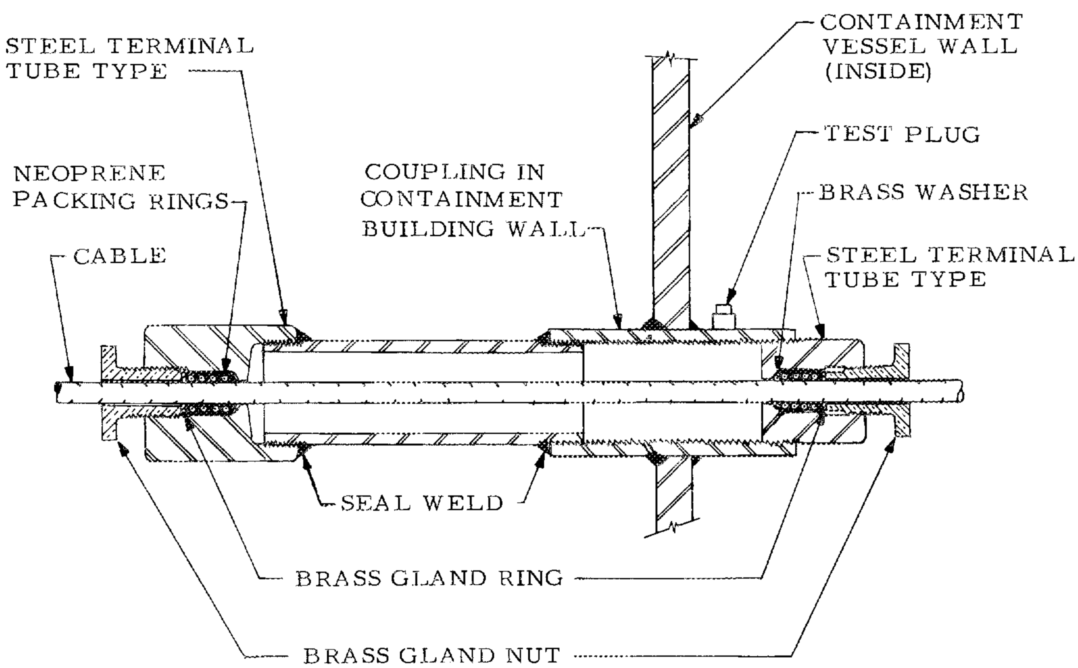



Fig. 9.24. Fermi Cable Penetration. (From ref. 1)

### 9.5.3.2 Potted

Potted penetrations utilize one or more layers of various types of sealing materials poured around the cables. This type of penetration may be shop fabricated or the potting materials may be poured in place in the field to permit the use of continuous cable. A prefabricated potted penetration assembly such as used at the Dresden and Big Rock Point plants is shown in Fig. 9.25 (ref. 8). Continuous-wire potted penetrations are illustrated in Figs. 9.26 (ref. 9) and 9.27. The first is typical of those used at the EBWR, while the second is used at the Peach Bottom plant.

Several different materials are often used to make the seal in this type of penetration. A nonshrinking cement grout is used to supply the necessary strength to enable the penetration to resist the design pressure. The "Chico" and "Chico A" materials* indicated on Figs. 9.25 and 9.27 are nonshrinking grouts that help to reduce cracking, voids, and separation from the penetration walls. However, since this material is relatively porous, nonporous epoxy is applied over the grout to form a leaktight seal. Glass beads or other fillers may be added to the epoxy

*Products of the Crouse Hinds Co.

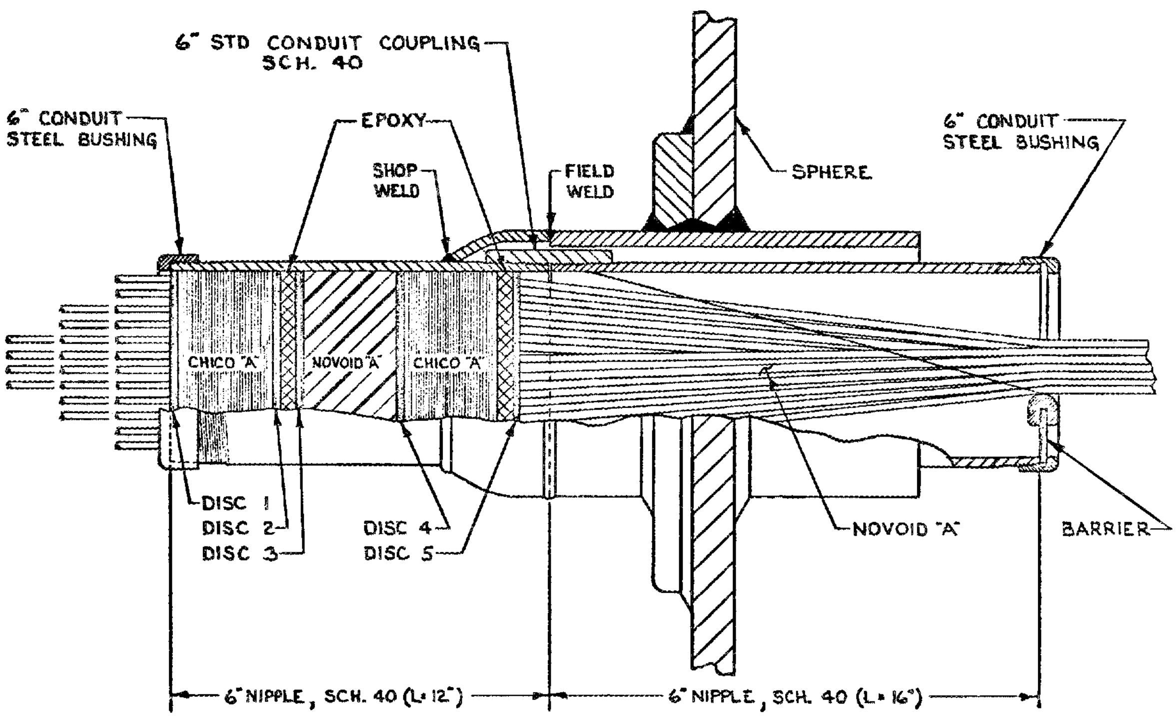



Fig. 9.25. Cable Seal Assembly Used at Dresden and Big Rock Point Plants. Chico "A" is a product of the Crouse Hinds Co.; Novoid "A" is a product of the G & W Electronic Specialty Co. (From ref. 8)

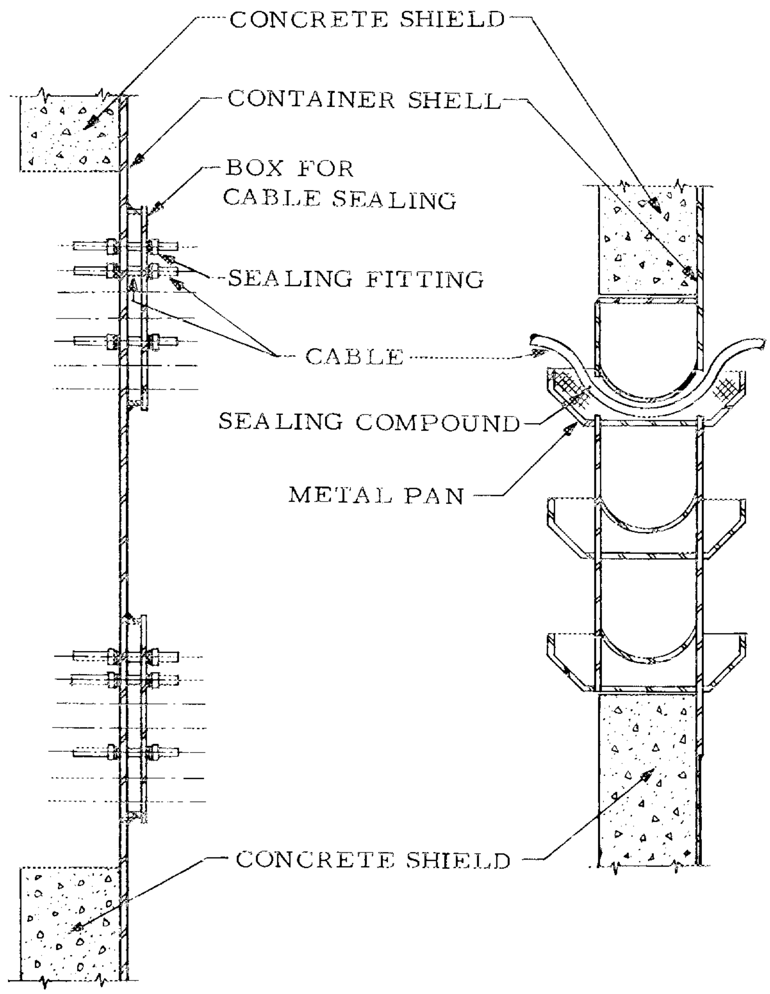



Fig. 9.26. Multiple Penetration Box and Continuous-Wire Multiple Conductor Electrical Penetration. (From refs. 9 and 10)

to reduce shrinkage. A third material, a pothead compound, is applied to seal any leakage paths that might form due to shrinkage of the epoxy. This material becomes more fluid when hot and thus would help to seal any leakage paths during high-temperature and high-pressure accident conditions. It is also flexible and will not crack due to temperature and pressure cycling. The "Novoid" and Novoid A" materials* shown on Figs. 9.25 and 9.27 are pothead compounds of this type. Since this material will flow very slowly even at room temperature, it either must be used on a horizontal surface, as in the U-tube of Fig. 9.27, or barriers must be provided as in Fig. 9.25.

The conductors or cables must be kept separated while the sealing materials are being poured. Several perforated disks are used for this

*Products of the G & W Electronic Specialty Co.

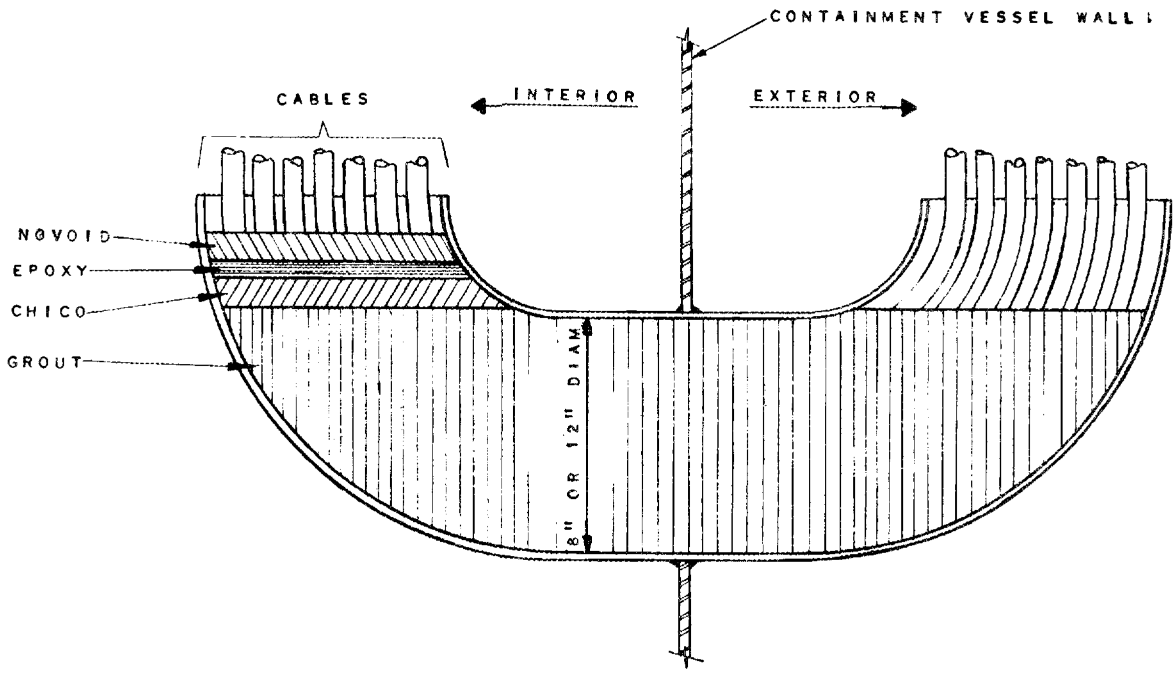



Fig. 9.27. U-Tube Potted Penetration Used at Peach Bottom. (From ref. 6)

purpose in the shop-fabricated penetration of Fig. 9.25. In the field-fabricated type, special care must be taken to keep the cables separated and to completely surround all cables with the sealing materials.

As with all continuous-wire penetration designs, cable sealed either internally during manufacture or with an additional potted seal must be used in the potted penetrations shown in Figs. 9.26 and 9.27. Another penetration that uses a sealing compound is shown in Fig. 9.28 (ref. 10).

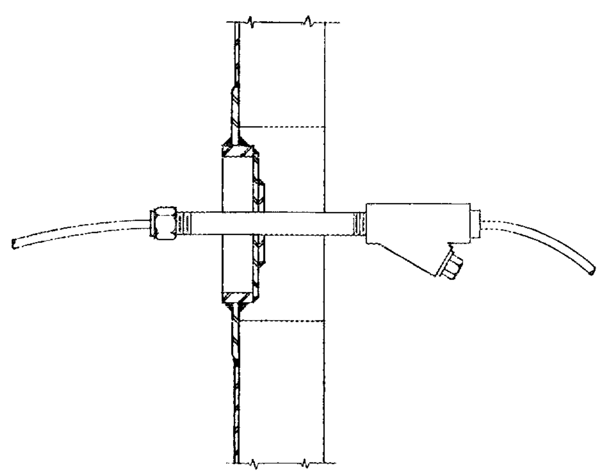


Fig. 9.28. EBWR Main Power Cable Penetration. (From ref. 10)

This is the design used for the three main generator cable penetrations at the EBWR. The cables were led through pipe sleeves welded to the containment vessel. The cables and the sleeves were then filled with a transformer compound, and the outer ends were blocked by filling a box built around the sleeves with epoxy resin. The ends of the cables were sealed to prevent internal leakage by dipping them into a rubber sealing cement. A multiple penetration box filled with a sealing compound to seal several penetrations at one time is shown in Fig. 9.26.

9.5.3.3 Gasketed

Gasketed seals are sometimes utilized in standard electrical devices, such as potheads, which may be used to seal some types of cables. The gaskets seal the various parts of the pothead assembly. A penetration of this type, used for 3-kv cables on the Peach Bottom plant, is shown in Fig. 9.29.

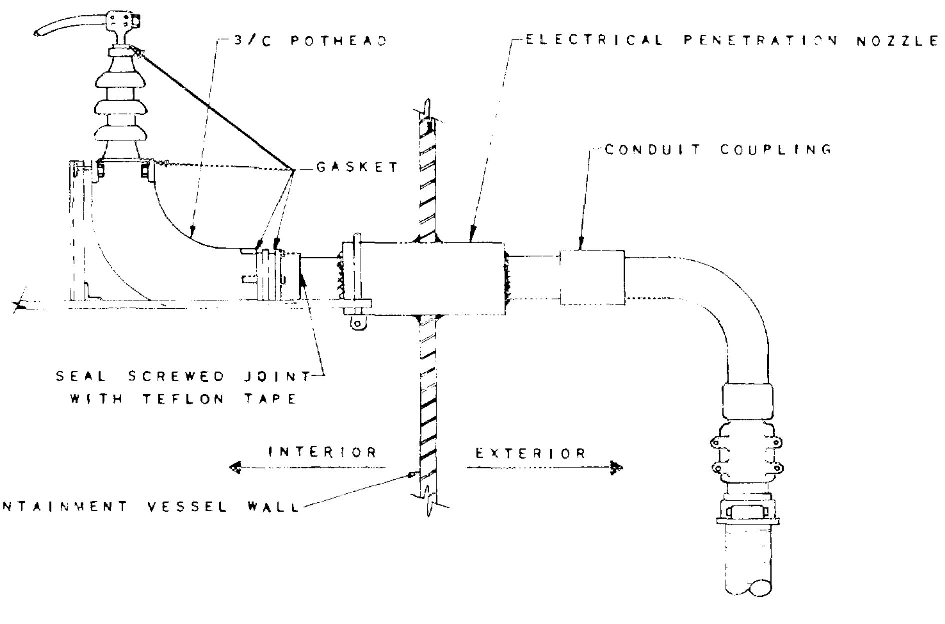



Fig. 9.29. Gasketed Cable Penetration Used at Peach Bottom. (From ref. 6)

9.5.3.4 Nonseparable Hermetic Connectors

One of the most significant sources of leakage on the NS Savannah has been gasketed-type electrical penetration assemblies.[11] A replacement program is currently under way using a new design developed jointly by the Scintilla Division of the Bendix Corporation and the Savannah technical staff. The new design incorporates a nonseparable hermetic seal by fusing solid electrical conductors into a steel membrane, using glass as the fusing material; it is similar to the separable hermetic seal hereafter discussed. The new design was based on service conditions of a 200-psi differential pressure and a 400°F temperature. Prior to installation, each fitting was proved to leak less than $10^{-6}$ cc/sec at a 100-psi differential pressure. Figure 9.30 shows the original and new designs.

A high-voltage bushing connector (Fig. 9.31) was utilized to pass each of the high-voltage circuits through the BONUS containment shell.[12] The connector employed was a standard, single-phase, 15-kv, gastight, fixed-stud insulator bushing mounted on a nonmagnetic plate. Attachment of the plate to the containment shell was accomplished by seal welding.

#### 9.5.3.5 Separable Connectors

Bulkhead-type separable connectors are frequently used for low-leakage penetrations, particularly for coaxial cable and for other conductors of relatively low power. Separable connectors rated for relatively high currents are also available. Assemblies are usually made up of several of these connectors mounted on plates, blind flanges, or pipe spools that are bolted to the nozzles through the containment wall.

Separable connectors are available as the so-called "environmental" type, which limits leakage to values on the order of one cubic inch of air per hour per connector at 30 psig, or for very low-leakage applications, the hermetic type may be used. The hermetic type limits leakage to less than one micron cubic foot per hour at one atmosphere. Examples of separable connectors used for containment penetrations are shown in Figs. 9.32 and 9.33.

### 9.5.4 Penetrations in Conventional Walls

Leakage around the outside of a penetration assembly is prevented in steel containment vessels because penetration nozzles are welded into the vessel wall and are pressure tested and leak tested as part of the vessel. In concrete containment structures with steel liners, the penetration nozzles are embedded in the concrete and can be seal welded to the liner. However, in other types of containment structures, such as low-leakage buildings of relatively conventional construction, other means of minimizing leakage around the penetrations may be required. Figures 9.33 and 9.34 illustrate possible penetrations for use in a steel panel wall and in an unlined concrete wall, respectively.

### 9.5.5 Testing

All electrical penetrations must be leak tested to demonstrate that they meet the specified leaktightness requirements. Penetration assemblies fabricated in the shop are shop tested with air, water, or steam before they are delivered to the field. Accident conditions can be better simulated and control can be maintained more easily in a shop test than in a field test. However, even with shop-tested penetration

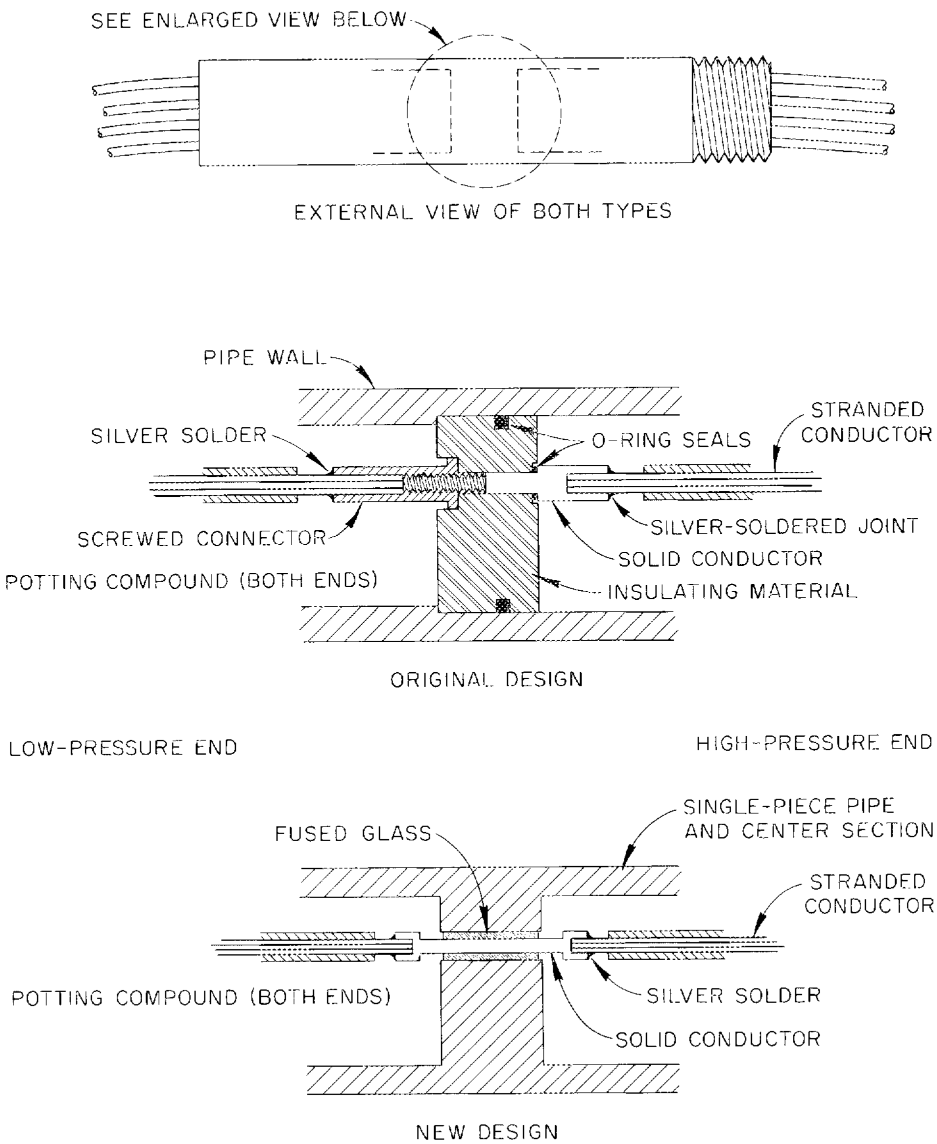



Fig. 9.30. NS Savannah Electrical Penetrations. (From ref. 11)

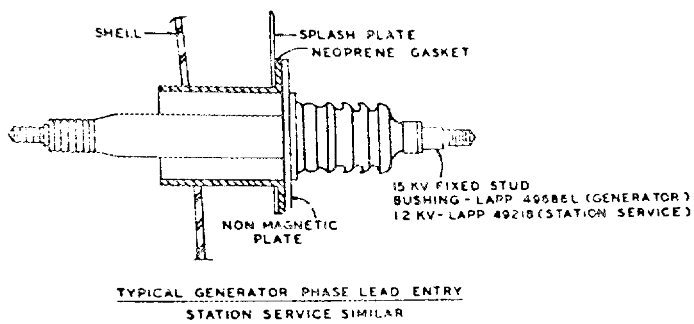



Fig. 9.31. BONUS High-Voltage Bushing Connector. (From ref. 12)

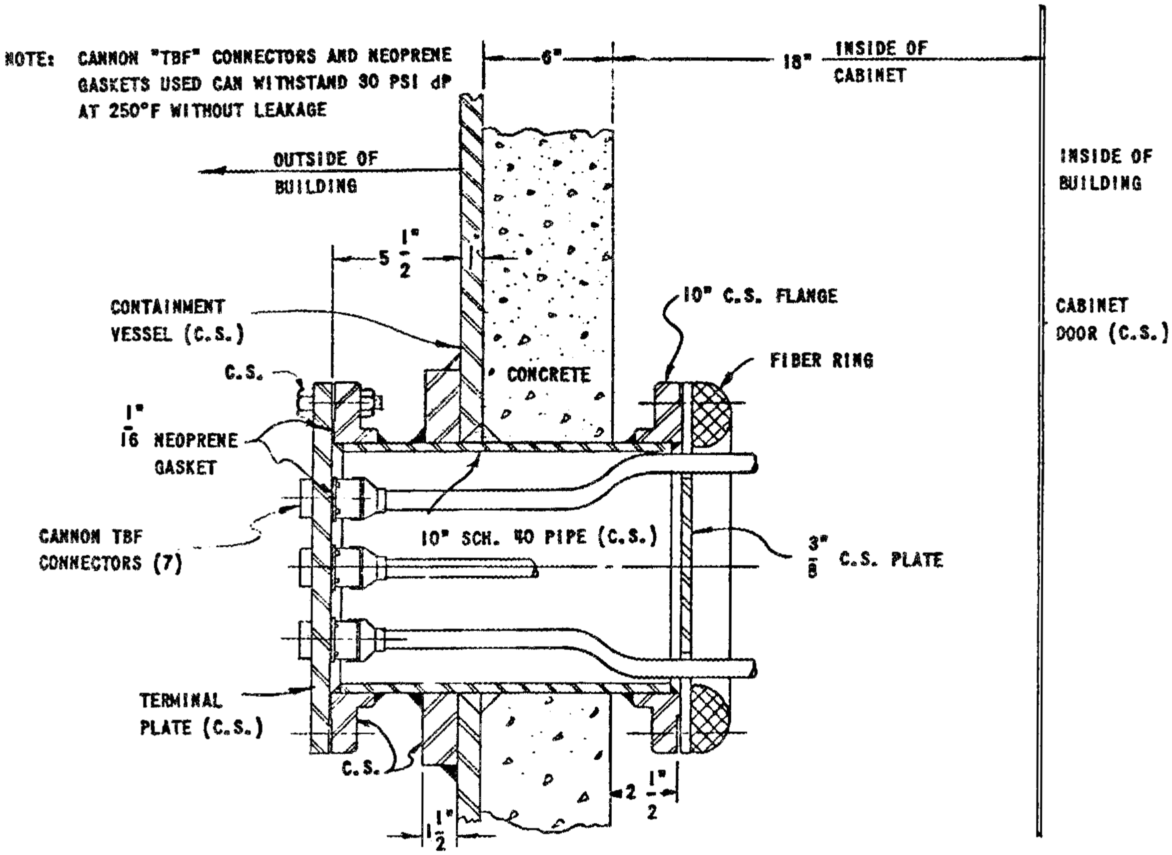



Fig. 9.32. EBR-II Electrical Penetration. (From ref. 1)

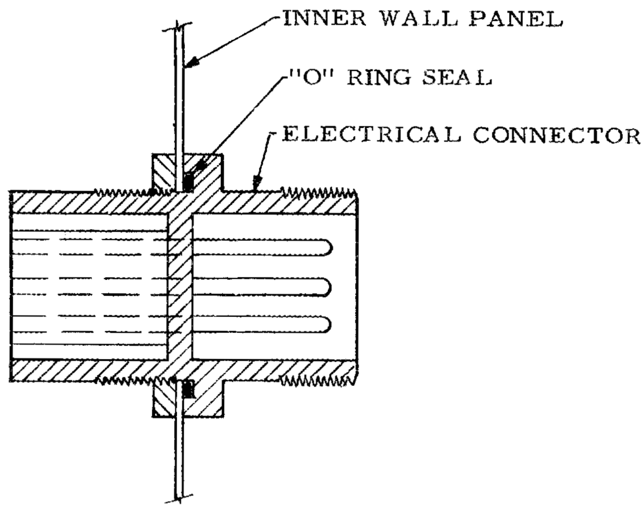



Fig. 9.33. Electrical Penetration Through Steel Wall Building Material Panel. (From ref. 2)

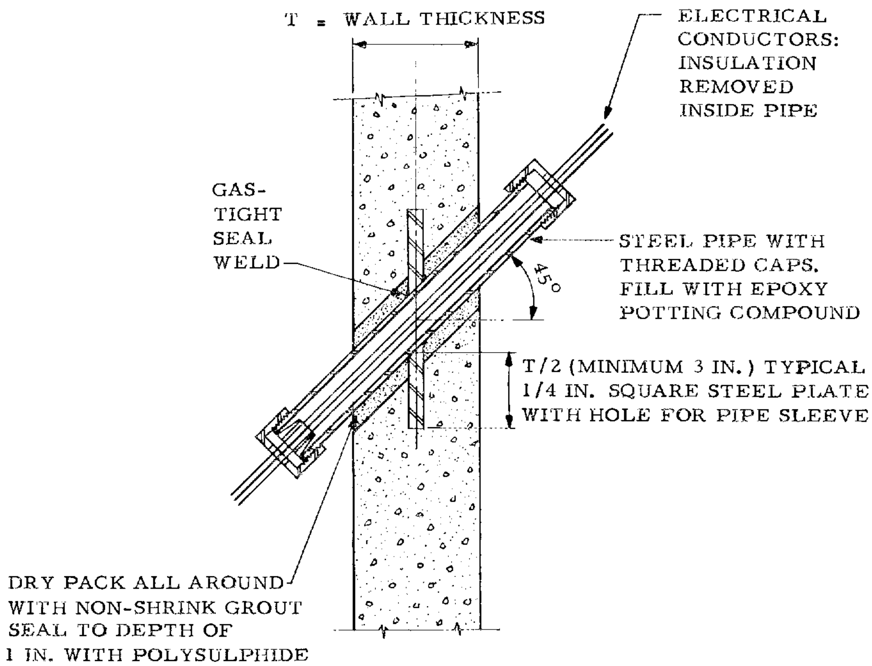



Fig. 9.34. Electrical Penetration Through Concrete Wall. (From ref. 2)

assemblies, field testing is required to demonstrate the leaktightness of the installed penetration. Field-fabricated penetrations, such as the continuous-wire shown in Figs. 9.26 and 9.27 can, of course, only be tested in the field.

During or just before the integrated leakage-rate test of the entire completed containment vessel, all penetrations are checked for leakage by a soap bubble test or other local leak detection test (see Chap. 10). Before the cables are connected to both sides of the penetration, installed penetration assemblies can also be leak tested individually, without pressurizing the entire vessel, by means of a vacuum box. In this method a box with sealing edges is placed over the penetration and pumped down to establish a differential pressure across the penetration. Leakage is detected by noting a rise of pressure within the box, by observing bubbles if there is a window in the box and the area has been covered with a soap solution, or by detection of a specific gas, such as a halogen or helium, which has been released on the other side of the penetration.

Some penetrations have been specifically designed to facilitate individual testing. The cable penetration shown in Fig. 9.24 is such a design. By pressurizing the space between the two penetrations, any leakage through either seal is detected by a decrease in pressure. This type of penetration is particularly attractive in that it eliminates the need for repressurizing the entire containment vessel each time the penetration is tested and would be desirable if frequent testing of these penetrations were necessary.

A small amount of leakage through an electrical penetration may be acceptable, but in practice any leaks detected in a local test usually are repaired, since a large number of even very small leaks could prevent the containment vessel from meeting its integrated leakage-rate requirements.

## 9.6 VENTILATION AND PRESSURE CONTROL SYSTEMS

### 9.6.1 General

The containment atmosphere must be controlled within appropriate temperature and pressure ranges during both normal and accident conditions. It may also need purging to remove radioactive materials or to restore or maintain normal atmospheric chemical composition prior to or during human occupancy. In some cases, a special inert atmosphere is required to limit or eliminate possible chemical reactions with reactive materials used within the containment system.

This section discusses various containment auxiliary systems and components used to control the atmosphere in conventional containment systems. Entire containment concepts based on a particular method of pressure control are also discussed briefly, but reference is made to Chapter 7 where these concepts are discussed more completely.

### 9.6.2 Ventilation Systems

The high-temperature components within the containment system are sources of heat that must be dissipated to the external environment in order to prevent the temperature of the containment atmosphere from exceeding that tolerable either by humans, if the containment structure is normally occupied, or by many components located within the containment system. Many insulation materials are not guaranteed at temperatures over 135 to 150°F, and the operation of many components is not guaranteed above some ambient temperature. Consequently, standard practice is to limit the operating temperature within a containment system to 120 to 135°F if possible.

Removal of heat is accomplished either by open ventilation systems, which may be rapidly closed during hazardous conditions, or by closed-loop heat exchange systems. In some cases, particularly where the ventilation system is also used to remove decay heat following an accident, a closed heating and cooling system may be used together with an open ventilation system. Open ventilation systems are the simplest and are used most frequently. Because of the possibility that the ventilation air may become contaminated with radioactive matter, means must be employed to maintain the activity in the effluent ventilation air at acceptable levels. This is usually accomplished by means of filters installed in the ventilation exhaust system. Closed heat exchange systems may be required if acceptable levels of radioactivity cannot be guaranteed using an open system. A closed system is used on the NS Savannah.

A closed heat exchange system may use forced circulation of the containment air past water-cooled heat exchangers located within the containment system. It is also possible to circulate the containment air outside the containment boundary to water or air coolers. However, using internal water coolers results in smaller penetrations and isolation valves.

Closed-loop heat exchange systems are also used for cooling certain components of the plant, such as radiation shields and pump motors. While not normally considered a part of the ventilation system, these cooling systems do remove a significant portion of the heat load that otherwise would have to be removed by the ventilation system.

In sodium-cooled reactor systems, the use of closed-loop water-cooled systems for transferring heat out of the confinement volume may not be permitted because of the possibility of a sodium-water reaction if a cooling line should rupture. This requires either that an open ventilation system with a suitable filter system be used or that the confinement atmosphere be circulated through isolation valves to coolers located outside the confinement envelope.

Controlled containment ventilation may serve any or all of the following functions, in addition to that of heat removal:

1. Supplying fresh air to occupied areas,
2. Controlling the flow pattern from clean areas to radioactive or potentially radioactive areas,
3. Maintaining differential pressure for the control of leakage,
4. Removing radioactive gases.

Fresh air supply facilities are required in all contained areas that can be occupied, including both those that are occupied during operation and those that are unoccupied during operation but which require purging prior to entrance after shutdown. These facilities usually consist of normal heating and ventilating equipment and control systems. The clean air is distributed internally by sheet-metal ducts or light-gage pipe ducts under slightly positive pressure. The exhaust air, because of its possible contamination, is normally drawn off under reduced pressure to prevent outward leakage from the ventilation system. The exhaust fans are usually located downstream from the containment air treatment facilities and just ahead of the discharge to the stack.

In addition to maintaining a reduced pressure in the exhaust system, it is preferable to keep a slightly negative pressure in the containment system to prevent outleakage through normal operation of the access locks, which would otherwise bypass the monitoring system.

The types of material and duct construction normally used are of three categories:

1. Standard sheet metal construction is used for air supply systems up to the containment penetration, for the distribution system inside the containment, and for the exhaust system up to the isolation valve or discharge penetration, where collapse of the duct during an accident would not violate containment integrity. Sheet metal ducts may be used for differential pressures in the range of 2 to 3 in. $H_2O$ above or below ambient.

2. Light-gage spirally welded pipe is used for pressures beyond those listed above in all types of supply, discharge, and exhaust systems. It is also used for leaktight low-pressure (less than 2 to 3 in. $H_2O$) systems that may convey contaminated air through occupied areas.

3. ASA standard line pipe is used where the ventilation system is part of the containment boundary and is subjected to full design pressure during an accident.

The portion of the piping welded to the containment vessel and those sections that constitute a part of the containment system must conform to the applicable code governing the containment vessel. Standard construction techniques compatible with the materials used are acceptable in most cases. Where severe radioactive contamination could occur, either special coatings or stainless steel may be required to facilitate decontamination.

Removal of radioactive contamination from the ventilation system air prior to exhausting it to the atmosphere, if required, is accomplished by filters or gas scrubbers incorporated into the ventilation system. These fission-product-removal systems are discussed in Section 9.7.

Efforts are normally made to control the moisture content of the containment atmosphere to prevent condensation on the inside of the containment vessel wall and on equipment located inside the vessel. This requires monitoring the dew point and keeping it below the equipment or containment surface temperature, either by using dehumidifiers in closed systems or by merely having sufficient ventilation flow in open systems. This is a relatively easy task unless a substantial steam leak exists. Conversely, an unusually high dew point may be an indicator of such a steam leak, and corrective action may be required.

### 9.6.3 Ventilation Stacks

The ventilation system exhaust in most nuclear plants is discharged to the atmosphere through a stack to provide dispersion of any radioactivity that might be contained in the exhaust air. The height of the stack is determined by the local meteorological conditions and by the degree of atmospheric dilution required. However, in most nuclear power plants, radioactive gaseous wastes from the radioactive waste disposal system are discharged to the same stack and the dispersal of these wastes determines the stack characteristics. The ventilation air then serves to provide a high flow rate in the stack and a high stack exit velocity, which increases the effective stack height and thus improves dispersion in the atmosphere. Methods of determining the degree of atmospheric dispersion under various meteorological conditions and with various discharge heights are outlined briefly in Chapter 4.

Conventional steel or concrete stacks are used in most nuclear plants, the shorter stacks frequently being made of steel and those above about 150 ft in height usually being concrete. Because the air discharged is at a relatively low temperature and is noncorrosive, special linings or coatings are not usually required. Stack characteristics at some representative nuclear power plants are listed in Table 9.2.

Table 9.2 Ventilation Stack Characteristics at Several Nuclear Power Plants

| Plant | Stack Height (ft) | Flow Rate (cfm) | Exit Velocity (fpm) |
|---|---|---|---|
| Dresden | 300 | 45,000–50,000 | 3000 |
| Humboldt Bay | 250 | 12,000 | 6000 |
| Big Rock Point | 250 | 30,000 | 3000 |
| Peach Bottom | 150 | 20,000 | 4000 |

### 9.6.4 Vacuum-Relief Devices

The containment pressure is not usually subject to significant variation during normal operation and may be controlled, in an open ventilation system, by controlling ventilation air flow either at the inlet or at the outlet of the ventilation system. Vessels not designed to withstand the negative pressure that might occur with faulty ventilation valve operation are normally equipped with vacuum-relief valves. The Peach Bottom containment vessel is protected from such negative pressures by a vacuum-relief valve with a weighted, nonmetallic pallet seat set to open at a negative gage pressure of 3.5 in. $H_2O$ (ref. 6). This device is illustrated in Fig. 9.35. The EBWR containment vessel was designed to withstand a maximum negative pressure of 0.5 psig and was provided with two vacuum-breaker valves to prevent negative pressures exceeding this amount.[10] Each valve is a 6-in. spring-loaded single-port regulator with a soft composition disk for tight shutoff. Each valve has a capacity of

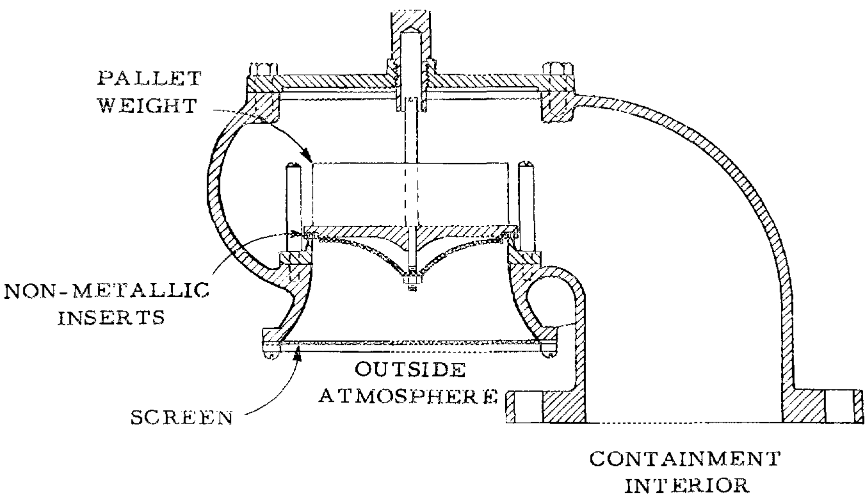



Fig. 9.35. Peach Bottom Vacuum-Relief Valve. (Varec, Inc.)

100 cfm, and each is set to open at 0.10 psig and to have full relief at 0.25 psig. Dual vacuum-relief valves are used in BONUS for backup protection in case one valve should fail to operate. Both valves are set to open at a negative pressure of 5 in. $H_2O$. At the containment design vacuum of 0.25 psig each valve will develop a minimum of 4000-cfm relief capacity. Either valve can accommodate the maximum expected barometric pressure change of 4 in. Hg per hour.[13] Vacuum relief devices are also used in the pressure-suppression type of containment to eliminate the vacuum that may exist in the drywell after an accident occurs.

The internal pressure in the NPR pressure-relief containment system (see Sec. 9.6.7) could be reduced below a negative pressure of 2 psig if the structure were initially filled with steam at 5 psig and the pressure-reducing fog spray system was turned on but failed to shut off at the proper time. Therefore, the structure is equipped with three 42-in. vacuum-relief valves that are set to open at a negative pressure of 1 psig.[14] Each valve has sufficient capacity to prevent pressures of less than −2 psig with the maximum fog spray flow of 6700 gpm condensing 5-psig steam in the building. These valves have rubber seats and are kept closed by a lever-and-weight arrangement. Negative pressure inside the building acts on the valve disk, which lifts the weight and admits air.

### 9.6.5 Inert Atmosphere Systems

Containment systems for reactors containing combustible materials, such as graphite moderator, sodium coolant, or combustible hydraulic fluids, may be required to maintain a reduced oxygen atmosphere to minimize the possibility of a fire. The reaction of such combustible materials with oxygen at the normal atmospheric concentration could add significant amounts of energy to the containment system during an accident and thus increase the required design pressure of the containment vessel. Reduction of the oxygen content of air to less than 5% would eliminate the explosion hazard with most organic materials and would reduce, by a factor of 4, the maximum oxidation reactions that could occur. This can be accomplished by purging the existing air with an inert gas, such as nitrogen, helium, or argon, or by burning the oxygen with a fuel to provide an inert atmosphere of nitrogen and carbon dioxide. This latter process also produces substantial quantities of water vapor, which should be removed, and carbon monoxide, which must be regarded as toxic and handled accordingly.

The NS Savannah, Hallam, and Peach Bottom reactor containment systems have inert atmospheres. In the NS Savannah a nitrogen atmosphere is maintained within the containment volume to prevent a fire if a failure occurred in the control rod drive hydraulic system and combustible hydraulic oil leaked out. The sodium-cooled Hallam plant uses a helium atmosphere at a pressure of 0.25 psig in the reactor cavity and a nitrogen atmosphere in the outer pipe tunnels.

In the high-temperature gas-cooled Peach Bottom reactor, the hot graphite moderator would react with any oxygen present following a main piping failure. Therefore, the plant is equipped with a propane burning nitrogen gas generator that produces nitrogen containing approximately 0.5% oxygen by volume within the containment vessel.[15] Makeup nitrogen is supplied by the gas generator, as required, to maintain the desired containment atmosphere.

### 9.6.6 Negative-Pressure Systems

Negative-pressure containment systems are used in conjunction with low-leakage conventional buildings in those cases where only a small amount of gas can be released from the reactor system in the event of an accident. The ventilation system is designed to maintain a reduced pressure within the building at all times so that all leakage will be inward. This system differs from other types of containment systems in that the ventilation system is not shut down in the event of an accident but continues to operate, or is started up if not previously operating, and depends on filters and other fission-product removal devices to remove radioactivity from the exhaust air. The air is exhausted up a stack to disperse any radioactivity not removed.

Systems of this type are used for the Hallam and Humboldt Bay refueling buildings to control the radioactivity that could be released if a fuel element were dropped and it melted during fuel-handling operations. At Hallam, the system provided also minimizes the possibility of sodium oxide being released if the radioactive sodium coolant is exposed to air and burns during refueling or maintenance operations.

Maintaining a negative pressure throughout the containment system to ensure that all leakage is inward requires that the ventilation system blowers have sufficient capacity and that the leakage paths and ventilation inlet are properly controlled and distributed. This containment concept is discussed in more detail in Chapter 7. Design details of low-leakage buildings are discussed briefly in Chapter 8. Exhaust filter systems are discussed in Section 9.7 of this chapter.

### 9.6.7 Pressure-Relief Systems

The pressure-relief type of containment was devised to permit the use of structures designed for relatively low pressures even when appreciable amounts of energy could be released during an accident. In this system, the initial release of high-vapor-pressure reactor coolant due to a primary system rupture is vented to the atmosphere, and then the vent is closed prior to the release of the fission products from the fuel by core meltdown.

This system is used at the Hanford NPR and at the Canadian NPD plant at Chalk River. Under normal operating conditions the light- or heavy-water coolant contains only a slight amount of radioactivity, the release of which does not constitute a serious hazard to the environment. The release of only the coolant and not the fission products is accomplished by the use of low-pressure valves or diaphragms in the relief vents that open or rupture to allow release of the flashed water vapor and by the use of one or more devices that seal the vents and prevent the release of fission products after the pressure has been relieved. In the NPD system a large glass diaphragm is broken to relieve the pressure, and a gravity-actuated flap gate closes the relief duct after a 10-sec interval.[16] The steam vent and closure device used in the NPR system is shown in Fig. 9.36. It consists of a steam vent cap, a butterfly valve, and a balloon-type backup closure. The cap is normally retained by a preloaded shear pin, which fails at 2 psig. After venting, the butterfly valve closes or, should it fail to operate, the balloon closure is inflated to provide a seal in the vent.[14]

A water or fog spray system is used in both the NPR and NPD systems to minimize the initial pressure buildup in order to prevent operation of the pressure relief system, if possible, to reduce the particulate matter contained in the vented gases, and to reduce the actual amount of water vapor discharge by condensing a portion of the steam. The spray system can also be used to reduce the pressure increase due to decay heat after the vents have closed.

The pressure-relief type of containment and the NPD and NPR systems are discussed in more detail in Chapter 7.

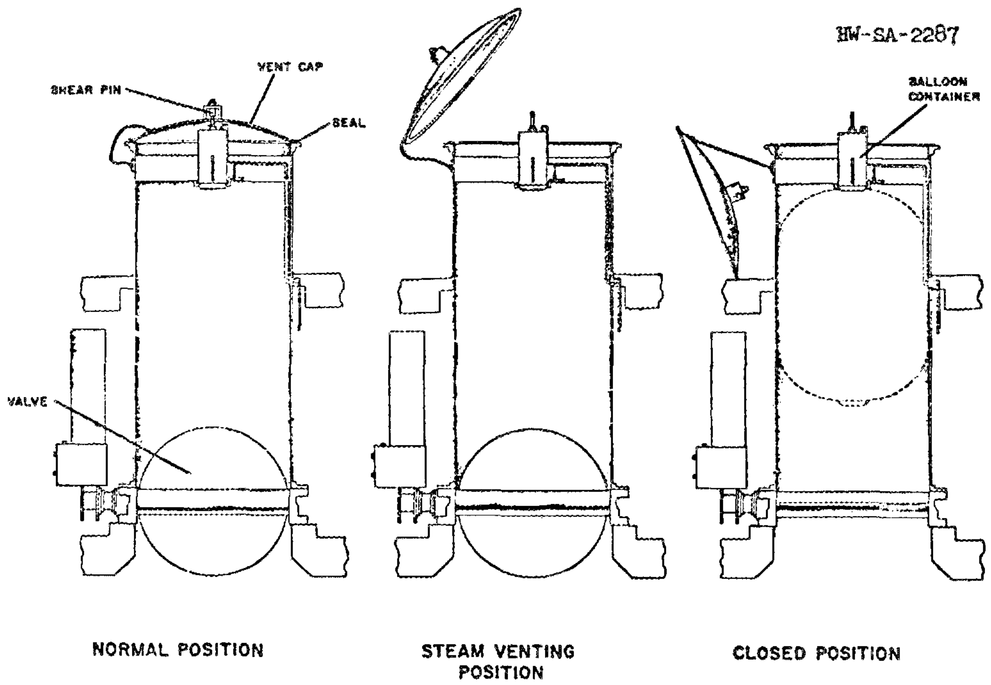



Fig. 9.36. NPR Steam Vent and Vent Closure. (From ref. 14)

## 9.6.8 Pressure-Suppression Systems

The peak pressure reached in a containment structure during an accident (i.e., the design pressure of the structure) can be reduced without venting the structure to the atmosphere if a rapid and reliable means of absorbing some of the energy released from the primary system is provided. Two different methods of using water to absorb energy from the reactor coolant have been applied to reactor containment for water-cooled reactors: pool pressure suppression and spray systems. A complete containment concept has been based on pool pressure suppression, and spray systems have been provided in many conventional steel shell containment structures as auxiliary systems.

#### 9.6.8.1 Pool Pressure Suppression

In pool pressure suppression, usually referred to as the pressure-suppression type of containment, the steam-water mixture resulting from a major primary system rupture would be vented directly to a large pool of water, which would condense the steam and keep the pressure from reaching its full unsuppressed value. Also, if any fission products were vented to the pool, the water would probably trap a significant fraction. Because the pool of water is always present and there are no valves or other devices that must operate, full reliance can be placed on the system functioning properly if it is needed.

This method of containment is discussed in Chapter 7.

#### 9.6.8.2 Water Spray Systems

Water sprays can be used as a means of intimately mixing cold water with the steam-water mixture released from a primary system rupture in order to condense the steam and reduce the pressure. This can be done either at the time of the accident to minimize the peak pressure or later to reduce the pressure within the containment structure and thus reduce leakage. The water spray should also be effective in removing fission products from the containment atmosphere after an accident and thereby reducing both the leakage of radioactivity out of the vessel and the direct radiation from the vessel (see Sec. 9.7.4).

As indicated in the preceding section, the NPD and NPR systems have water spray and fog spray systems, respectively, to minimize the pressure peak that will occur following a major primary system piping failure. In addition, the NPD system also incorporates a limited spray system to douse small steam leaks.

Several, more conventional, containment systems also include post-incident cooling systems to accelerate the reduction in the containment pressure and to wash down contaminants. Dresden and Big Rock Point both have several circumferential headers of spray nozzles mounted inside the vessel near the top and supplied by the fire system pumps. Outside elevated water tanks provide BONUS with both inside and outside water spray systems and fire protection, thereby averting dependency upon a source of power for pumping. At Big Rock Point, the feed line, which is a branch line from the core-spray postincident cooling system, is normally dry, but its supply valve is automatically opened on high containment vessel pressure after a 15-min time delay. This delay, which may be overridden to provide earlier manual operation, is provided to allow the operator time to override a possible spurious signal. Each set of sprays has a total capacity of 400 gpm and, if operated for approximately 24 hr, would raise the water level in the containment vessel to near grade. Further addition of water would have to be stopped, since a water level higher than 3 ft above grade would overstress the vessel shell. Therefore, intermittent operation is proposed after the vessel pressure has been reduced to approximately atmospheric in order to conserve the vessel capacity for prolonged accumulation of spray water. Extended operation could be accomplished by withdrawing water from the spray system drains and recirculating it through an external heat exchanger supplied to cool the core spray and containment spray water.

The water supply for the spray system may come from an elevated tank located inside the containment boundary, as in the EBWR containment vessel, or from an outside pumped supply such as the fire system. An internal supply may be considered more reliable, since it would not require a power source for pumping and would not require penetrating the containment boundary. However, it is still necessary for the proper signals to be actuated and for valves to operate to start the spray flow. It is usually not practical to supply sufficient water from an internal tank to adequately cool a large plant.

Operation of a spray system, unlike pool pressure suppression, requires electrical and mechanical components to function during accident conditions, and containment systems based upon full reliance on such a system have not yet been used. All conventional containment systems built to date for which spray cooling has been provided have been designed for the maximum pressure resulting from a primary system rupture without spray cooling. The sprays have been provided only to reduce the consequences of an accident below those for which the plant was designed. However, the effectiveness of a reliable spray system in reducing containment pressure should approach that of a pool pressure-suppression system. Where reliance is placed on sprays to prevent containment rupture, an assured power supply for the required pumps is a necessity.

## 9.7 FISSION-PRODUCT REMOVAL AND TRAPPING SYSTEMS

### 9.7.1 General

Most containment systems have filters and other means of contaminant removal to trap radioactive materials and to remove them from the containment ventilation exhaust. Some types of containment, negative-pressure systems in particular (see Sec. 9.6.6), rely on such devices to remove radioactive materials from the ventilation exhaust during accident conditions. Conventional containment systems, which are designed to be isolated (not ventilated) during an accident, may employ fission-product removal systems to remove small amounts of radioactivity from the ventilation air during normal operation and may depend on these or other accident-adapted systems (with recirculation in some cases) to clean up the air discharged from the containment system after a fission-product release has occurred. Many conventional closed containment systems also include provisions for removing large quantities of fission products from the containment atmosphere after an accident in order to reduce both the direct radiation from the containment vessel and the inventory of fission products available for leakage out of the vessel. This is accomplished by means of a closed-loop filtration system totally contained within the containment vessel.

The selection of the types of fission-product removal devices to be used and their design for specific applications depend on many factors, including the physical and chemical forms of the materials to be removed, their concentrations and total quantities, the removal efficiency required, and the environment in which the devices must operate. Removal of a contaminant from an air stream or a volume of air is based on one or more of the physical or chemical properties of the containment. Particulate materials are usually removed from an air stream by passing the air through pleated filters. The type of filter to be used depends upon the size of the particles to be filtered and the corrosive nature of the air. However, high-efficiency filters with a rated efficiency of 99.97%, based on 0.3-μ-diam particles, are usually specified when radioactive particulate material is to be removed. Reactive gases may be removed by passing the air stream at the proper temperature over sufficient surface area of a material with which the gas will chemically react. Less reactive or inert gases may be adsorbed onto activated charcoal or other adsorbent material. Liquids in the form of droplets or mists may be removed in much the same way as are particulates, using filters that are especially resistant to moisture and other corrosive conditions, or they may be dissolved in a larger volume of water. Because the fission-product removal systems frequently used in containment systems must be able to cope with a wide range of possible contaminants and operating conditions, their design may utilize any or all of these and other removal methods.

Most of the experience with removal of fission products from air streams has been accumulated at nuclear process laboratories and reprocessing facilities where large volumes of highly contaminated air are handled routinely. Filter systems have also been used extensively to remove small amounts of particulate contamination from large volumes of ventilation air in all nuclear plants and laboratories. However, filters and other fission-product removal devices have been operated much less, or not at all, under the high-pressure, high-temperature, and high-humidity conditions that would exist within a containment system as a result of a serious accident. Thus, the effectiveness of systems designed for these accident conditions can, in many cases, only be estimated until further testing is carried out under properly simulated conditions.

The designs of many fission-product removal devices are based upon designs of similar gas-cleaning equipment used in nonradioactive applications. Where the physical and chemical forms of the contaminants are similar, similar operating conditions should result in essentially the same removal efficiencies in both the radioactive and nonradioactive cases. However, removal of fission products often requires a greater removal efficiency with a smaller inlet concentration than is common in normal gas-cleaning operations, so suitable industrial gas-cleaning experience on which to base the design of a fission-product removal system may not be available. Furthermore, fission-product removal may present some special problems not present in most gas-cleaning applications, such as assuring that the decay heat generated by the fission products is adequately removed, either by sufficient air flow or special coolers, and providing sufficient capacity for large quantities of nonradioactive or slightly radioactive materials, in addition to the smaller quantities of fission products of principal concern, which may load the removal devices. Minor duct leaks may offset high filter efficiencies, and therefore fission-product removal ducts, housings, etc., may require improved designs compared with those of most conventional gas-cleaning installations.

The types of fission-product removal devices in general use in containment systems are discussed briefly in the following sections. These are principally particulate filters and iodine-removal devices for use in exhaust air streams. Removal of radioactive noble gases, such as xenon, krypton, and argon, is possible but is not usually attempted because the higher allowable concentration of these materials results in fewer disposal problems and because the removal systems that would be required are complex and costly; therefore, such systems are not discussed. Methods proposed for removing fission products from the containment atmosphere after an accident are outlined.

## 9.7.2 Particulate Filters

Several types of particulate filters are commercially available for a wide variety of applications. A typical filter system for removal of radioactive particulate material from an air stream includes prefilters in series with high-efficiency filters. An air stream with a high moisture content may also require a moisture separator to protect the filters from damage or clogging by water droplets or by moistened dust. If the filters are subjected to high operating temperatures or to incendiary materials, filters and filter frames must be noncombustible and heat resistant. A more complete discussion of various types of filters and other equipment for removing particulate materials from air is presented in ref. 17.

### 9.7.2.1 Prefilters

Prefilters are used as roughing filters upstream of high-efficiency filters to capture large dust particles and prolong the useful life of the more expensive high-efficiency filters.

The most commonly used type of prefilter consists of a glass fiber medium in a heavy paper-fiber frame. The filtering medium usually is a 2-in.-thick pad of interlaced glass fibers having good dust-holding properties. Several other prefilter designs are shown in Fig. 9.37 (ref. 18).

Prefilters 24 x 24 x 12 in. are considered standard for use in filtration systems with a capacity of 1000 cfm or greater. Such filters have an initial pressure drop of about 0.15 to 0.35 in. $H_2O$ static pressure, depending upon the type of prefilter selected.

### 9.7.2.2 High-Efficiency Filters

High-efficiency filters, when required, are installed downstream of the prefilters. Filter efficiencies vary from a 99.50% minimum for high-temperature ceramic media to a 99.97% minimum for the standard glass-asbestos media. The efficiencies are based on tests utilizing an aerosol

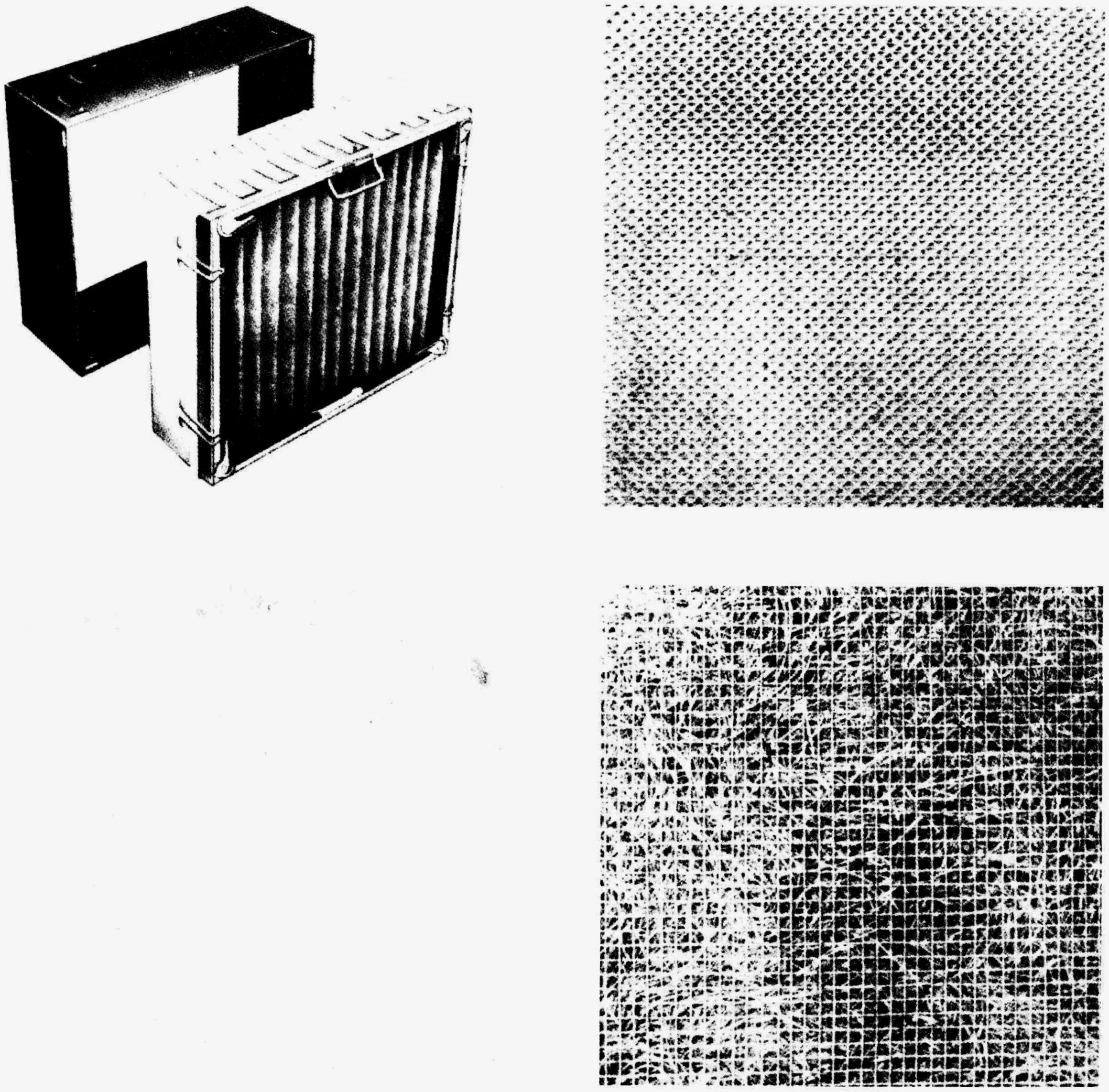


Fig. 9.37. Various Types of Prefilters. (From ref. 18)

of dioctyl phthalate (DOP) of 0.3-μ-diam particle size. The DOP test methods are discussed in Section 9.7.5. The standard high-efficiency filters used for the removal of micron and submicron particulates from an air stream have pleated glass or glass-asbestos media with aluminum, asbestos, or heavy kraft paper separators. The pleated filter medium is contained in a 3/4-in. plywood frame measuring 24 x 24 x 11 1/2 in. Figures 9.38 and 9.39 illustrate typical high-efficiency filter units. Cellulose-asbestos filter paper constitutes a fire hazard, and its use is not generally recommended.[19] The 24 x 24 x 11 1/2-in. high-efficiency filter is rated at approximately 1100 cfm, with an initial pressure drop of about 1.0 in. $H_2O$ static pressure. Since various types of filters, filter media, separators, and frames are available in various sizes and efficiencies, reference should be made to the technical literature for additional information before attempting to specify filters for a particular installation.

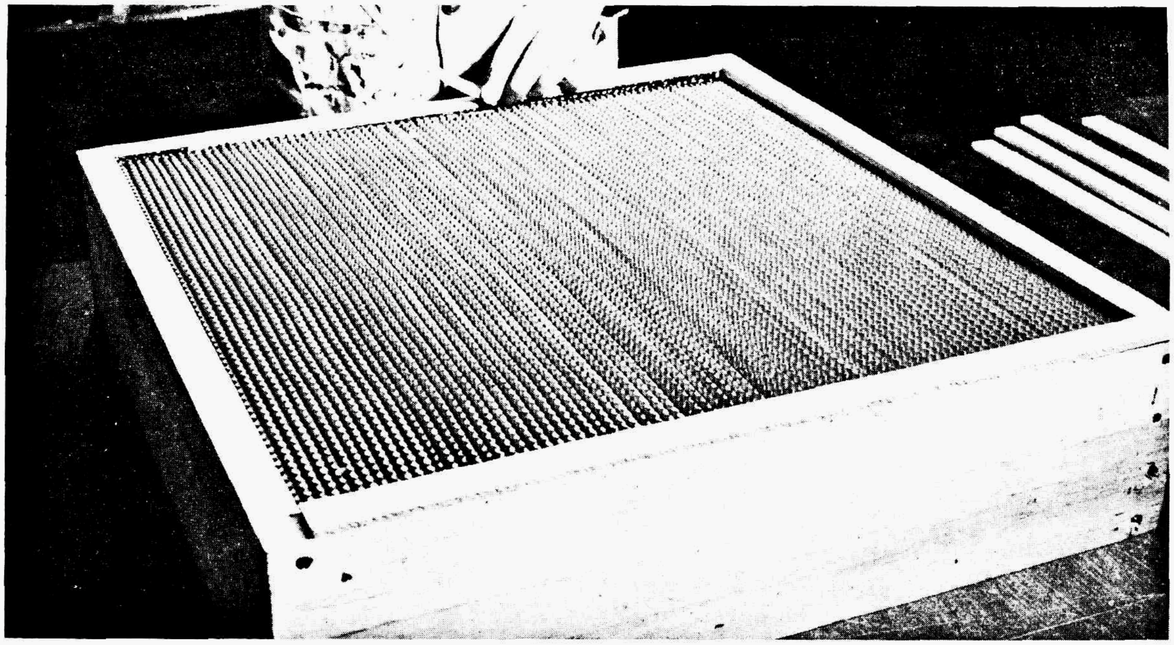


Fig. 9.38. Asbestos-Cellulose High-Efficiency Filter. (From ref. 17)

#### 9.7.2.3 Moisture Separators

If a filter system is to be used to filter the steam-air mixture that could be released during a major accident in a water-cooled reactor plant, a moisture separator may be required instead of or in addition to a prefilter. A moisture separator is used upstream of the the normal filter system to reduce the clogging effect that would otherwise occur with the sudden flow of wet steam or fog through normally dusty filters. Excessive moisture could also weaken and cause rupture of absolute filters. Moisture separators may also be used in conjunction with charcoal beds in an effort to prevent loss of efficiency caused by excess moisture.

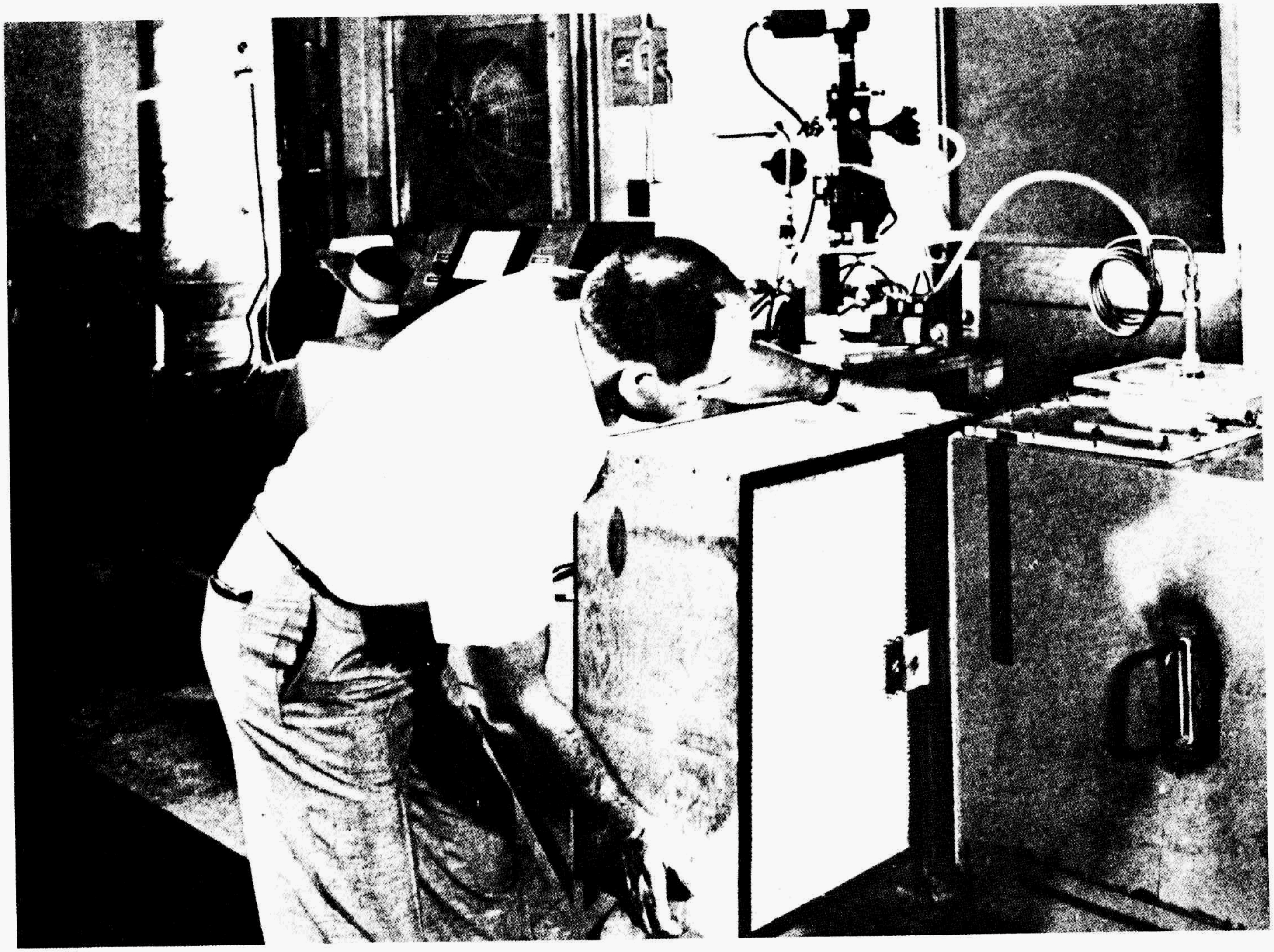
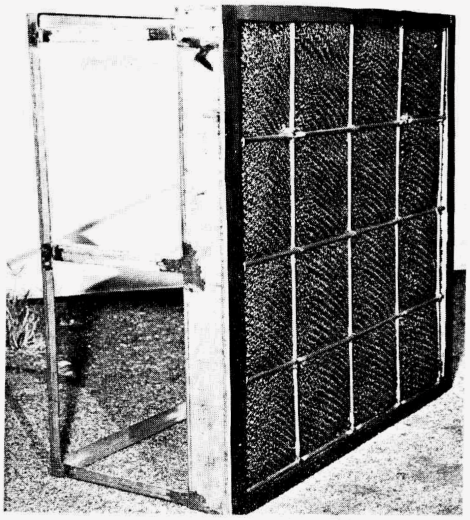


Fig. 9.39. High-Efficiency Filter Unit. (From ref. 18)

Favorable results have been obtained from experimental work on one type of moisture separator for this application.[20] The moisture separators tested consisted of 2-in.-thick mats of Teflon yarn wrapped over stainless steel reinforcing wire. A unit of this type is shown in Fig. 9.40 that is rated at 400 scf/min. $ft^2$ at a 0.95-in. $H_2O$ differential air pressure. During the tests, glass-asbestos fiber filters subjected to a flow of wet steam and fog without upstream moisture separators quickly became almost completely plugged with liquid water, and some rupturing occurred. With moisture separators, performance of the filters was satisfactory.



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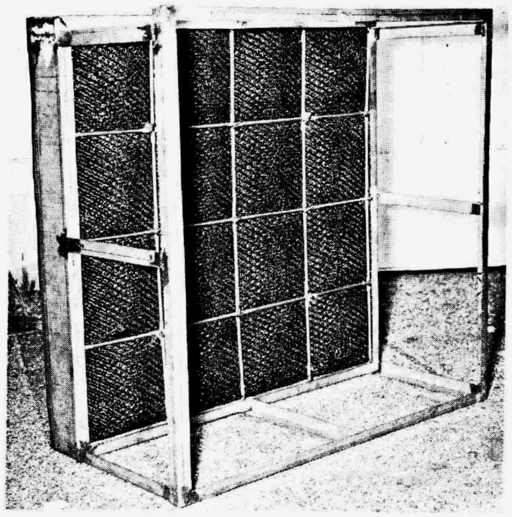

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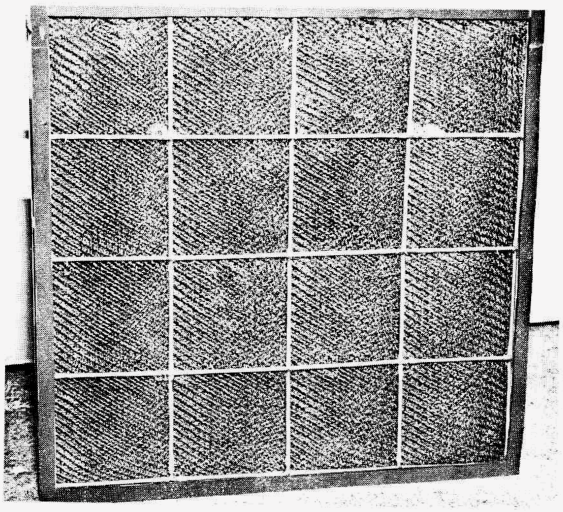

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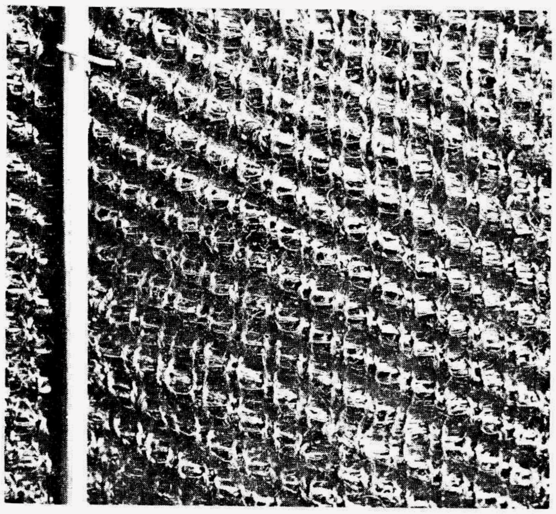

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Fig. 9.40. Moisture Separators. (From ref. 20)

## 9.7.3 Iodine-Removal Devices

Radioactive iodine is of particular importance because of the extremely low concentration in air that is tolerable and may be discharged from a nuclear plant both during normal operation and in the event of an accident. Considerable attention has therefore been given to methods for removing iodine from air streams,[21,22] particularly at reprocessing facilities where radioactive iodine is released routinely when spent fuel is dissolved. These methods include liquid absorption in scrubbing columns, adsorption on certain solid materials, cryogenic applications, or chemical reaction. Since the form of released iodine is determined by environmental conditions at the time of release, the application of these methods might differ from the application used in a chemical processing plant. The same methods also remove other radioactive gases of less biological significance.

Iodine-removal systems can be designed for nearly any application and desired removal efficiency by using multiple units and combinations of methods, if necessary. A difficulty with some methods is that of assuring that the effectiveness of the system will be maintained after long periods of idleness, as in the case of systems installed in containment vessels. Periodic testing and replacement is desirable and may be required. Testing of these systems is discussed in Section 9.7.5. The principal methods used for radioactive iodine removal are discussed below.[23,24]

### 9.7.3.1 Caustic Scrubbers

In a caustic scrubber, an aqueous solution of NaOH is pumped over the packing of a scrubbing column through which the contaminated air is passed. Iodine, as well as any acid contamination, is removed from the air stream. Iodine-removal efficiencies of 95 to 99% can be obtained. A caustic scrubber used in the containment building of the Oak Ridge Research Reactor, see Fig. 9.41 (ref. 25), was tested at an air flow rate of 5000 to 6000 cfm and an iodine concentration of 15 $\mu g/ft^3$; it achieved an iodine-removal efficiency of 99%. The removal efficiency is relatively independent of air flow rate over quite a wide range but increases with inlet iodine concentration. In general, caustic scrubbers are useful for reducing the molecular iodine concentration by one to two orders of magnitude; however, iodine adsorbed on particulates may also be removed by action of the fluid.

### 9.7.3.2 Heated Silver Nitrate Beds

A method developed and used principally at Hanford for removing radioactive iodine from the gaseous effluent of a Purex facility uses silver nitrate supported on packing in a column. The beds are maintained at 375 to 400°F. With a gas velocity of 1 ft/sec, removal efficiencies of 99.9 to 99.99% were obtained in development tests, and the full-scale units operate with efficiencies of 99 to 99.9%.

### 9.7.3.3 Silver-Plated Wire Beds

In another method, the gas stream is passed over silver metal plated on a wire mesh. A knitted mesh of silver-plated copper ribbons and silver-plated stainless steel wire mesh have both been used with success in tests.

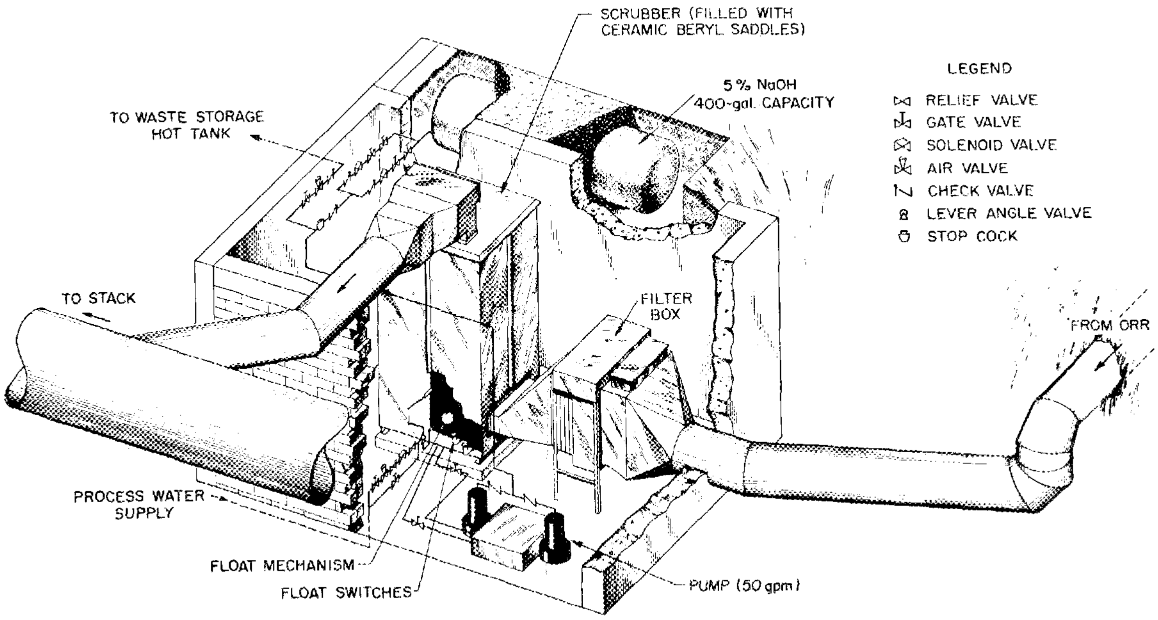


Fig. 9.41. Oak Ridge Research Reactor Caustic Scrubber. (From ref. 25)

Iodine-removal efficiencies of 90 to 99.9% have been obtained for concentrations of 0.05 to 700 mg of iodine per cubic meter of air, respectively. Efficiencies increase somewhat with temperature and with moisture content of the air. The pressure drop through the beds is relatively small compared with that of other methods. There is indication that the removal efficiency would decrease with use much more with this method than it would with activated charcoal beds.

### 9.7.3.4 Activated Charcoal

Activated charcoal has been used widely for radioactive iodine removal. Removal efficiencies of 99.99% and greater have been achieved in activated charcoal beds, and large amounts of iodine can be removed with little loss of efficiency. Efficiency decreases somewhat with increasing temperature, as well as with water vapor in the air stream.[26] However, this material presents a greater combustion hazard than the other iodine-removal materials. The pressure drop is greater than for silver-plated wire mesh but is less than for other materials with removal efficiencies above 99%. Either petroleum of cocoanut-shell charcoal may be used in activated charcoal beds, the latter being the more efficient for iodine absorption. The honeycomb unit shown in Fig. 9.42 was designed to hold particles of activated charcoal or other iodine-removal materials, as well as to remove particulate material. Results of tests on the iodine-removal efficiency of activated carbon in various forms are presented in ref. 27.

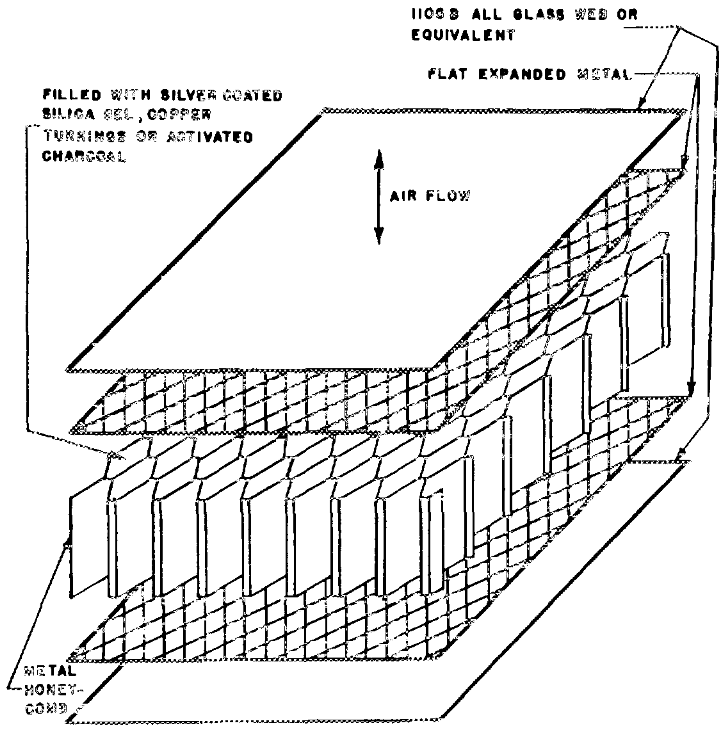


Fig. 9.42. Diffusion Board for Iodine and Particulate Removal. (From ref. 18)

### 9.7.4 Containment Atmosphere Clean-up Systems

Various systems have been proposed to remove fission products from the containment atmosphere following a major accident. This would reduce the direct radiation dose from the containment vessel and thus reduce the exclusion radius or amount of containment shielding required. It also would reduce the amount of radioactivity that could leak out of the containment vessel and further limit the environmental consequences of an accident. Little information is available concerning the effectiveness of such systems, so the credit that can currently be taken for them in assessing the consequences of an accident may be much less than their actual value.

#### 9.7.4.1 Recirculating Filter Systems

A filter system essentially the same as that used in the exhaust system of a conventional containment vessel and in the exhaust from a negative-pressure, low-leakage conventional type of building may be used in a closed, recirculating ventilation system located entirely within the containment vessel. The filter units may be enclosed within a radiation shield so that the collected fission products are not radiation sources. The system must be designed to operate in the accident environment, and the electrical power supply to operate the blowers must be reliable under accident conditions. Supplemental cooling of the filter beds may be required if the air flow through them could become insufficient to cool them. Since the air is recirculated, filters with an efficiency of less than 99.97% (a high-efficiency filter) may be suitable for removing the radioactive particulate material, depending upon the time and decontamination level requirements.

### 9.7.4.2 Spray Systems

Spray or deluge systems are provided in many containment systems. While their primary purpose is to reduce the pressure resulting from an accident, as discussed in Section 9.6.8, they should also be effective in removing particulate and water-soluble fission products from the containment atmosphere and the containment interior surfaces. The radioactivity washed out of the air and off the surfaces may be simply collected in the bottom of the containment vessel where it would not present a direct radiation source, or the water may be passed through filters that would remove the bulk of the radioactivity. The water for the spray system may be stored in an elevated tank within the containment vessel or it may come from an outside source.

The effectiveness of a water spray in washing fission products out of air may be approximated from an analogy with the "rain-out" of radioactivity released to the atmosphere. Methods of estimating this effect are presented in ref. 28, and data on the effectiveness of sprays in removing iodine are presented in refs. 29 and 30. However, even with good experimental data on washdown, uncertainties arise because of lack of information on the types and physical forms of the radioisotopes that will be encountered in the containment atmosphere. This subject is discussed further in Chapter 4.

### 9.7.4.3 Fog Systems

Fog systems have been proposed to wash radioactivity out of air more effectively than water sprays. Because the fog particles are much smaller than the drops in a water spray, they can interact better with fine particulate matter and can provide greater washing effectiveness with less water. A fog system may not be as effective as a spray in condensing steam and reducing containment pressure, however. As with the spray system, good quantitative data on the effectiveness of such a system are not available.

### 9.7.4.4 Foams

It has been suggested that in the event of an accident, a containment vessel could be filled with a stable foam that would trap the fission products and prevent them from leaking out of the vessel. This is probably a more positive means of trapping fission products and reducing leakage than those mentioned above, but it does not substantially reduce the direct radiation from the containment vessel. It also could present more difficulties in cleaning up after an accident.

### 9.7.5 Testing

Fission-product removal systems should be periodically tested in place to be certain that their design efficiency is actually being achieved. The efficiency of an installation that contains high-efficiency filters depends not only on the integrity of the individual filters but also on the integrity of the entire installation and, in particular, on the proper installation of the filters in the mounting frames. The efficiency of such systems should be measured by in-place tests after each filter change; however, routine in-place testing is also necessary for maintaining the integrity of operational installations. Some fission-product removal schemes, such as sprays, can only be tested for operability; their effectiveness is usually estimated from the results of laboratory tests. Testing requirements and test methods employed on the usual type of fission-product removal systems are discussed in the following paragraphs and in Chapter 2.

#### 9.7.5.1 Particulate Filters

At an air-cleaning seminar[31] at Harvard University in June 1957, one manufacturer of high-efficiency filters alleged that at least one manufacturer was using an efficiency-measuring instrument that needed calibration. On the basis of that remark and evidence that seemed to bear out the allegation, the AEC, in cooperation with the Army Chemical Center at Edgewood, Maryland, tested random samples from filter stocks of atomic energy plants. The survey disclosed a significant percentage of unsatisfactory filters. As a consequence of these findings, the AEC established two quality-assurance stations to inspect and test new high-efficiency particulate filters for the atomic energy program. The General Electric Company at Richland, Washington (HAPO), was designated as the station for installations west of the Mississippi River, and the Chemical Corps Arsenal, Edgewood, Maryland, provided service for locations in the east.[32] Effective January 1, 1963, the Oak Ridge Gaseous Diffusion Plant of the Nuclear Division of Union Carbide Corporation was designated an AEC quality-assurance station to replace the Chemical Corps Arsenal, Edgewood, Maryland.[33] At these filter testing centers, new filters are inspected and tested for efficiency according to MIL-STD-282.[34] While this service was placed on a voluntary basis for participants in the atomic energy program, its use is urged by the AEC.

Use of the service offered by the quality-assurance stations asures the receipt of sound, efficient filters, but the efficiency of a filter system can be determined only by a suitable in-place test. The Oak Ridge National Laboratory has modified the DOP testing technique used by the quality-assurance stations for such in-place testing of filter installations.[35,36] The procedure is as follows:

1. A polydisperse aerosol of dioctyl phthalate (DOP), produced by atomization with compressed air, is discharged into any convenient air intake ahead of the filter bank.
2. The concentration of the unfiltered aerosol is measured by drawing a sample from a point ahead of the filter bank and passing it through a forward light-scattering photometer.

3. The concentration of the aerosol in the filtered air is measured downstream of the filter bank (frequently after the exhaust blower to assure the withdrawal of a thoroughly mixed sample).

4. The efficiency of the filter bank is calculated from the unfiltered and filtered concentration measurements.

In order to conduct this test, there must be a sufficient length of duct between the point where the aerosol is introduced and the filter bank to allow thorough mixing of the aerosol-air mixture. When the length of duct is too short to provide adequate mixing, alternate procedures are available; these are discussed in ref. 36.

If the in-place test indicates an unsatisfactory efficiency, it is usually possible to find the source of the trouble by probing the downstream side of the filter bank with the photometer probe. It is not unusual to find leakage paths in the filter housing or in the gasket areas, as well as through holes in the filter medium. Experience at the Oak Ridge National Laboratory has indicated that it is relatively easy to achieve a system efficiency of 99.97% or better for small single and double filter systems, but relatively few existing large multifilter installations were found with efficiencies this high when initially tested. However, it was found that new systems of all types have a good chance of passing an in-place test if close attention is given to details of design and fabrication and if the filters are installed under competent supervision.

The NS Savannah used a similar procedure for routine in-place testing of the reactor compartment ventilation system filters.[37]

## 9.7.5.2 Iodine-Removal Systems

One method of testing an iodine-removal system requires that a small sample of radioactive iodine be injected into the inlet air stream of the installed system and that samples of the inlet and outlet streams be taken and measured for iodine concentrations.[23] This method is subject to several potential errors, however, and care must be taken when determining iodine-removal efficiency. In one commonly used method, the inlet and outlet gas samples are passed through small sample collectors, such as activated charcoal or caustic scrubbers, and it is assumed that the efficiencies of the two sample collectors are the same. The collection efficiencies of materials, including the materials in the device being tested, vary with the iodine concentration, as well as the chemical and physical form, which may change during transport between sampling stations. In the case of reduced sample collector efficiency, an overestimate of the iodine-removal efficiency would be made due to the lower concentration in the outlet gas. Another possible error results when the radioactive iodine appears as a true gas or is chemically adsorbed on particles of dust small enough to penetrate the collectors.

The NS Savannah procedure for the routine in-place testing of the iodine-removal efficiency of the reactor compartment ventilation system filters[33] is outlined below:

1. The ventilation system is set to operate smoothly at rated test capacity.

2. Personnel observing and conducting the test wear activated charcoal respirators and protective clothing. All other personnel evacuate the compartment that surrounds the containment vessel.

3. The iodine-injection system is connected to the intake of the emergency ventilation system. The compressed air supply and air-metering system are connected to the iodine-injection system.

4. Activated charcoal iodine-sampling units are installed in ports upstream and downstream of the iodine-absorption system undergoing test.

5. Iodine air monitors are set for operation in the compartment and other appropriate locations during the iodine test.

6. The activated charcoal sampling units are set into operation.

7. The iodine-injection device is activated by first starting an air flow over the gas ampoule containing the iodine and by then crushing the outer walls of the stainless steel tubing to break the ampoule. Health physics monitoring is required during this operation.

8. Operation of the test is continued for a period of 2 hr (subject to change depending upon results of initial iodine penetration test). The air flow through the injection device and sampling units is then stopped, the sampling units are removed for radioassay, and the injection device is removed to a radioactive storage area.

### 9.7.5.3 Requirement for Filter Testing

Those plants that require highly efficient filtration and iodine-removal systems to reduce the release of radioactivity to tolerable limits should require routine in-place testing of these systems to demonstrate their effectiveness. While no standards exist for conducting in-place tests, tacit approval has been given to the method of testing high-efficiency filters that employs air-operated aerosol generators. The results of numerous tests have been used and accepted as the basis of review by the Commission for licensing purposes (see refs. 35 and 36 and Chap. 2).

There are no entirely satisfactory alternatives to performing in-place tests of high-efficiency filters and iodine filters. Care in the design and fabrication of an installation, plus care in the installation of filters, do not necessarily assure having a system of adequate efficiency; however, they increase the probability significantly.

## 9.8 OPERATION OF CONTAINMENT SYSTEMS

C. S. Walker*

As indicated in previous sections of this report, the purpose of containment is to limit the release of radioactivity to acceptable values following any credible accident. The methods of achieving this objective rely not only on the containment enclosure itself, but also on the operation of equipment, such as isolation valves that must be closed, air

*Oak Ridge National Laboratory.

locks that must not have both doors open simultaneously, spray systems where needed to prevent unsafe rises in pressure or temperature or both, filters to reduce the radioactivity of any effluent being released via a stack, and blowers to direct the effluent through the filters and stack. This equipment, together with the necessary sensors, amplifiers, interconnections, relays, control valves, etc., must operate successfully as an integrated system if the containment enclosure is to fulfill its function.

In nuclear processes, systems that are designed for protecting the reactor and the public are called safety systems.[38] Much attention has been given to the system that decreases the reactivity if the reactor power exceeds a safe value. Generally, a substantial number of safety rods are provided so that a complete failure of any one device will not significantly decrease the safety potential of the shutdown system. Similarly, several redundant channels of instrumentation are provided so that an instrument breakdown will not render the system inoperative. In addition, the outputs of the instruments are often combined, using coincidence techniques, to minimize the effects of a single danger signal, presumably false, which otherwise would initiate an unnecessary shutdown. The instruments used in this service have been highly refined and are built to more rigid, precise specifications than are analogous devices in, for example, chemical plants. The safety actuators (rod drives) are also designed with great care because of the exacting requirements of this safety-system service. All of this care and effort is for the purpose of securing the reliability required of a protective system. In the general sense, a protective system is any system having the design objective of limiting the results of a credible accident to acceptable values. Thus the system composed of the equipment and instruments needed to achieve containment is in reality a protective system.

### 9.8.1 System Reliability

It might be argued that since the containment system will probably not be needed during the life of the plant, the containment closure system may be designed with less reliability than is required for nuclear systems. This reasoning is false. High reliability is required wherever the failure of a system of protection has been determined to be unacceptable.[39] The containment enclosure is provided, often at great expense, to protect the general public from the consequences of an uncontained, but credible, reactor accident. Correct action by the containment closure system is essential to achieving containment, and it is therefore essential that the system operate reliably.

The hope is often expressed that a statistical treatment of component failure experience will result in a quantitative expression for system reliability. If all system failures resulted from component failures, and if all component failures were random events, such a quantitative expression would become a possibility.[40,41] A numerical rating is not possible at this time, however, as attested by the substantial number of nuclear safety system failures that have resulted in reactor core damage or destruction where component failures had not occurred.[42-44]

The correct operation of the reactor containment system will be determined largely by the skill of the designer in meeting the frequently conflicting objectives of serviceability and safety. Containment closure systems cannot be absolutely perfect, and failures will occur. Failures which cause safety actions that interfere with the normal operation of the plant will reduce the serviceability of the entire facility, and by definition, failures that prevent corrective actions to be taken in an emergency will impair safety.[45] The first type of failure may be called a safe failure, or failure to safety; and the second type of failure may be called an unsafe failure, or failure to danger. The fail-safe principle, which can never be completely realized in practice, is an attempt to improve safety by arranging for the more common types of failures to be failures to safety.[41] Failures to danger, unlike failures to safety, are not self-detecting and can be revealed only by a failure-detecting monitor or a periodic test. Unsafe failures prevent "fail safe" from becoming a reality.

Serviceability may now be defined as the propensity to be free from safe failures, and safety as the propensity to be free from unsafe failures.[46] Were it possible to assign numerical values to serviceability and safety, reliability would be some function of these values. Reliability, then, is a combination of serviceability and safety.

There are two general classes of instrumentation and equipment systems, each employing a separate order of reliability, with separate degrees of serviceability and safety. Control systems are for the purpose of operating the plant in the manner intended by its design, and the consequence of a failure of a control system is at most a shutdown of the plant, either directly because of the failure or indirectly because of the subsequent action of a safety system. The reliability of a control system will therefore be based on economic considerations, with serviceability an objective. Protective systems, on the other hand, are for the purpose of providing protection where a failure to protect would result in damage to the reactor plant, public jeopardy, or damage to the program. Thus the reliability of protective systems must be of an order higher than that of control systems, with safety being the overriding consideration.

Protective systems should be capable of providing protection in spite of the failure of any single portion of the system that has a credible probability of failing. Examples of such parts of a system are relays, amplifiers, sensors, switches, connectors, interconnections, contact matrices, and actuators and their associated devices. There have been instances in which design errors have allowed protective systems to be incapable of performing as required, and the monitoring or testing procedures as applied have failed to disclose the deficiency.[42,45,47] It should be recognized that failures can occur in portions of the system that have not been identified as vital and which, therefore, lie outside the area of surveillance and test. It should be a design objective to search out the possibilities for and then minimize the occurrence of such failures.

The minimum reliability of a containment system should be of the highest order, since the need for containment may be the result of the failure of other systems that were designed to have high reliability. Also, the containment system may be the only protective system capable of functioning, as for example, following a fuel loading or unloading accident when the reactor is shutdown. It is mandatory that physical and functional independence be maintained between the containment system and other plant protective systems, as well as between the containment system and all control systems.

### 9.8.1.1 Protective Channels

A single channel for containment system actuation consists of the sensor, amplifiers, connections, and the actuator and its associated devices arranged so that a given process variable, having exceeded preset limits, can cause the equipment comprising the channel to take the correct action required to achieve or maintain containment. In such a simple protective channel, not interconnected with other channels, the instrument channel includes the sensor, amplifiers, and connections but does not include the actuator and the associated devices controlled by the actuator; however, all the considerations of channel testing and isolation must apply to the actuator and its associated devices, as well as to the instrument channel, since such equipment is a vital part of the containment channel.

Redundancy may be achieved by employing two or more independent and isolated channels, either alike or unlike, to accomplish the same given objective. The designer then has the option of interconnecting the channels so as to cause each input to affect any or all of the actuators of the complete containment closure system. It should be a design objective that these interconnections introduce a minimum impairment of channel isolation. When this type of interconnecting network is employed, each instrument channel is considered to terminate at the interconnection. As in the case of a simple single channel, all the considerations of testing and isolation must apply to the actuators and controlled equipment lying beyond the interconnection.

There should be a minimum of two functioning instrument channels associated with each given process variable used in the system. These instrument channels must be isolated physically and be independent in operation. When coincidence is used and system testing depends on coincidence, the failure of one of three instrument channels should require a shutdown of the plant by manual procedures if the failure prevents further testing and if repair of the failed channel cannot be completed in a time comparable with the maximum allowable interval between tests.

The employment of diversity will decrease the likelihood of a single event disabling both or all safety channels. The inputs associated with a given process variable may consist of two unlike measuring techniques; for example, fluid flow may be determined by means of a venturi or by the differential pressure across a pump. Also, a containment closure system may use two unlike protective channels to achieve the same end result. As an example, a valve closure channel and a series charcoal bed may be paired where the failure of either has a reasonable probability and fission-product release as a consequence of failure is unacceptable. The independence of the valve closure channel and the charcoal bed must be maintained, and the operability of both must be established by a testing or monitoring procedure.

9.8.1.2 Testing and Monitoring

Either of two methods may be used to determine the operability of the containment system. In the first method, the system is provided with built-in monitoring or testing equipment to allow the operability of substantially all of each channel to be verified during plant operation. Remaining portions of the safety channels are then tested during plant shutdown. The second method employs surveillance during operation and complete channel testing during shutdown. The first method is applicable to plants having long operating cycles, whereas the second method is applicable only to plants having short operating cycles. Neither method alone produces a protective system that is inherently more reliable than the other.

Monitoring utilizes measurements made during normal operation of the plant to determine whether a protective channel is capable of carrying out its design function. No disturbances are introduced into the channel at any point. The process is carried out by a continuous sensing of certain important internal variables in a protective channel, with alarm indications if any of these parameters should vary beyond a pre-established limit. For example, the monitoring of a vacuum tube in an amplifier might consist of indications and alarms showing whether the plate current and cathode voltage were each within their normal operating limits, and cables and connections to ion chambers can be monitored to determine whether the circuits are completed. Monitoring cannot be applied to all devices; for example, no measurements can be made on a relay during normal operation to determine that it will function properly on demand.

Where built-in monitoring is employed, it should be a design objective that unsafe failures be detected, since safe failures are self-detecting. Where a choice exists in arranging a given component to fail in the safe direction or to fail in the unsafe direction, it is acceptable in the interest of serviceability to adopt the unsafe failure provided the unsafe failure is detected through monitoring. When coincidence is employed, unsafe failures that are detected should cause the failed channel to trip. When coincidence is not employed, an unsafe failure detected in one of three or more channels should cause an alarm; an unsafe failure detected in one of two channels (or detected in one of two remaining channels in a multichannel system) should cause the containment system to be activated.

By definition, testing requires that one of the parameters of a protective channel be varied sufficiently to determine whether the channel responds correctly. Unlike monitoring, testing can allow an examination of the variations in the protective channel parameters. On-line testing may require that the actuators and associated devices cannot be allowed to complete their normal response to a signal requiring containment. However, the test should include as much of the system as possible, that is, it is desirable that the flow or temperature be perturbed so as to include the sensor and, where possible, the final protective device, such as a valve, should be actuated.[45] In cases where the test is less extensive, provision should be made for surveillance of the untested portions. There should be no interference with the corrective operation of the protective channel during the test; that is, the test signal should supplement or be superimposed on the normal signal. In addition, the test should simulate as faithfully as possible the actual behavior of the process variable. The frequency with which the tests are performed should be related to the predicted or experienced failure rate.

The system employed for testing should be arranged so as to constitute a minimal breach of isolation of the separate protective channels. If the system for testing is common to otherwise separate channels, the layout must be such that there is no increase in the probability of a failure in one portion of a channel causing all channels to fail. Failures originating in the test equipment should not affect the safe operation of more than a single protective channel.

Surveillance may take the form of observations of instrument response to system perturbations or the observation of recorder charts during normal operation to detect abnormalities. Surveillance should be employed wherever built-in testing or monitoring facilities are not provided or are incomplete.

Protective system faults that are not detectable by monitoring, by on-stream testing, or by surveillance should be detected by tests performed during shutdown periods. Periods of operation between such shutdown tests should be related to the predicted or experienced failure rates. Where it is necessary to disassemble a channel or portions of a channel to conduct these tests, the proper reassembly of the channel must be verified.

### 9.8.1.3 Independence of Channels

The predicted reliability based on the assumption of the random occurrence of faults cannot be achieved in practice because of the failure to obtain and maintain channel independence.[48–51] Multiple or simultaneous failures can be the result of either of two causes. First a design or maintenance error may result in all channels being incapable of performing as required. Faulty test methods may allow this condition to remain undisclosed. Second, there may be insufficient channel isolation, with the result that all channels are subject to simultaneous failure as the result of a single event. It is acknowledged that since the same environment must to some degree be shared by the individual channels, they may be subjected simultaneously to wetting, heating, or other factors leading to failure. Diversity may be employed to good advantage in obtaining channel isolation by the use of redundant channels of different construction or type.

The maximum isolation between plant control systems and the containment system is required. Any failure of a control system that may induce an accident requiring containment must not be allowed to cause a failure of the containment system also. Similarly, isolation between the containment system and other plant protective systems is required. The sharing of sensors, or any portion of protective channels, by the containment system and any other system should be prohibited. Special surveillance should be given to situations where sensors and other vital components of separate systems might be wetted, or overheated, or otherwise caused to fail by a single event.

It is acceptable for signals to be extracted from a containment system for the following purposes:

1. Alarm in order that the operator may take appropriate action,
2. Automatic corrective action employed with the objective of forestalling a plant shutdown provided the results of such an action are in the direction of safety.

In order to prevent a single error in maintenance or a single physical failure from impairing containment actuation, careful attention should be given to the routing of the electrical conductors of different channels. Where both control and protection are effected by the same process variable, the conductors to the sensors of these control and protective channels should not occupy the same conduit or raceway. Even in those cases wherein the process variables are different, but the action of a given protective channel will be needed as the result of a failure of a control channel, the conductors to the sensors of such control and protective channels should not occupy the same conduit or raceway. The conductors to sensors of a given process variable comprising a redundant group of protective channels should not occupy the same electrical conduit or raceway. Conductors for sensors of unlike process variables may occupy the same conduit or raceway except where these process variables are grouped to form a single redundant group. Physical separation should be maintained between the channels of a protective system where proximity might reasonably be expected to lead to simultaneous failure.

The layout of instruments and cables for protective systems should be such as to provide ready identification of these systems with respect to less vital control systems. Where the correct operation of a protective system is dependent on the correct operation of switches, relay contacts, or similar devices usually associated with plant control, these devices should be identified as part of the protective system and should be subject to all of the requirements of a protective system.

Where routine alteration of set points of routine disarming of a protective system is necessary and where protection would be lost because of failure to switch to the correct set point or to rearm the protective system, the reliance on administrative control over these operations constitutes a breach of channel independence. If instrument ranges must be switched in order to provide protection over the full range of operation, one of the following methods should be utilized:

1. Manual switching of ranges with trip points established at both high and low levels such that if ranges are incorrectly switched the channel is tripped,
2. Automatic reset of the trip point of each of the redundant channels, either continuously or in step fashion, with the reset being accomplished in each channel by independent instrument channels each capable of computing and controlling the reset,
3. Manual reset of the trip point of each of the redundant channels, typically in either of two settings, with the reset being permitted in each channel by independent instrument channels, each capable of monitoring the correctness of the setting.

When it is required that a protective channel be bypassed routinely, for example, during each reactor startup, it should be a minimum requirement that the bypassing of each of such channel be annunciated. A better method is to arrange also for the bypassing of each channel to be permitted by a separate instrument channel only when conditions are such that it is safe to bypass.

A protective channel can be disabled in a rather simple way by an erroneous setting of its trip level. As a minimum, the required trip level established for each channel should be displayed clearly, and the actual setting of the trip point should be displayed clearly or should be readily ascertainable. The required trip level and the actual setting could be displayed by signs affixed to the front panels of the equipment. Whenever adjustments, not necessarily accessible to the operator, are capable of putting an instrument on the wrong range or otherwise disabling an instrument channel, such adjustments should be appropriately limited or guarded.

### 9.8.2 Operational Requirements of System Components

#### 9.8.2.1 Valves

Whenever containment is contingent upon a sealed enclosure, some means must be provided to reliably seal each penetration. Electrical connections may be sealed permanently, but the seals on pipes and ducts must take the form of isolation valves that remain open during normal plant operation but which must be closed to make the containment effective.[38] These isolation valves should be controlled to close automatically whenever containment is needed, and interlocks should prevent their being reopened until conditions make it safe to do so.

The requirements of the containment system are, in at least one sense, more stringent than those of the nuclear safety system. The nuclear process and its control may usually be taken as a single problem, and therefore the provision of a multiplicity of safety rods gives safety in numbers. Increasing the number of rods decreases the effect of the failure of any one of them to release, and the effect of such single failures can be rendered unimportant in this way. The containment system may, however, be required to block a multiplicity of pipes and ducts. Here, there is danger in numbers, since the failure to close any exit can violate the containment. The analogy to multiple safety rods lies in multiple valves to block each line. Provision of valves per line in numbers comparable to safety rods could be very expensive, especially for the case of large ventilating ducts where the extra safety is most needed. Since increasing reliability by buying safety in numbers is uneconomic, it is clear that the relatively small number of valves actually provided in each line must be as reliable and trouble-free as possible.

The operating system for containment closure should be designed so that the failure of any single portion of this system cannot prevent the achievement of containment.[39,52] In general, two isolation valves in series in each line will be required; however, a single valve may be sufficient in case an alternate arrangement has been provided, as for example, the valve and charcoal bed discussed in Section 9.8.1.1. Independence of the valve and alternate device is required, as discussed in Section 9.8.1.3; and the operability of both the valve and alternate device must be established by testing or monitoring, as discussed in Section 9.8.1.2.

It is clear that containment valves are required to close when they are needed. Typical containment enclosure leakage-rate specifications range from 0.1 to 2% per day, and even a relatively small valve, which by failing to close may provide a path directly between the enclosure interior and the atmosphere, will leak at a rate very much larger than specified for the whole system. The case for valves in water and steam lines is not so clear. In some cases these lines lead to closed systems whose integrity may exceed that of the containment enclosure. Under normal conditions, transport of radioactivity into these closed systems does not constitute a hazard to the general public. However, the auxiliary system might not be in its normal conditions, and in any case it is desirable for fluids that might be highly contaminated to be confined to the containment volume.

In a once-through air-ventilation system that normally exhausts through a stack, on the occurrence of an accident, valves are required to close in the intake and exhaust ducts of the system. If the valve in the intake duct failed to close, any excess pressure in the containment enclosure could be vented directly to the atmosphere through the open intake port. The containment would thus be violated, with activity release perhaps taking place at or near ground level rather than through filters and up the stack. It is worth noting that it might be impossible to close such a valve by direct manual operation because of the high radiation level resulting from the release.

Accidental closure of an isolation valve when it is not needed for containment may also have undesirable consequences. As an example, an accidental closure of the intake or exhaust valves discussed in the above paragraph would interrupt ventilation. This might result in overheating of some plant components or in exposing personnel within the vessel to radiation from airborne activity that would normally be exhausted up the stack. Accidental blocking of steam lines, feedwater lines, or cooling-water lines could compromise major items of plant equipment. By appropriate design it has been possible to make some plants operate without outside ventilation. For these plants, of course, the problem of accidental closure of ventilating duct valves does not occur. In many reactor power plants, energy is transferred from the reactor to an external turbine-generator by a hot fluid in a pipe that penetrates the containment enclosure. Therefore, the closing of the containment valves would, at the least, interrupt power generation. Despite the possible undesirable results of an accidental closure of any one or all of the isolation valves, the plant should be designed to allow such safe failures to be tolerated. Otherwise, the conflict between serviceability and safety leads to a dilemma having no practical solution.

The actuators to close isolation valves in an emergency must be capable of meeting stringent requirements in an adverse environment. For example, the reactor accident that requires containment may produce missiles or release steam or water or other fluids in such quantities that the integrity of electrical circuits and air lines and the operability of electrical equipment may be in jeopardy. Also, a failure of pneumatic or electric power could have necessitated the containment closure. Therefore, the probability of electric or pneumatic power being available is at a minimum at the instant the isolation valves must be closed in an emergency. Thus, the closure of isolation valves by means of stored energy should be a design requirement.

Another example of a valve vital to achieving containment is the vacuum-relief valve. Most containment vessels are built to withstand a positive internal pressure and are not braced for a negative internal pressure. Relief valves are therefore provided so that outside air can enter if needed to avoid a condition of excessive vacuum. Such valves must meet all the rigid operating requirements of isolation valves, and in addition it should be noted that a failure to relieve an internal vacuum may be as catastrophic as a failure to close the containment. In any case, a leaking relief valve may be as great a hazard as a leaking isolation valve. The requirements of reliability discussed in section 9.8.1, including independence, testing, and effects of single failures, must be satisfied.

Check valves offer certain advantages, as well as disadvantages, compared with valves that require energy to close them. No external actuators or instrumentation are required. However, positive seating with very low leakage may be more difficult to assure than with valves having external actuating mechanisms. At best, such valves have a limited application, probably only in lines carrying liquids where actuation and leakage are less of a problem than in lines carrying gaseous fluids. If the valve is held open by the normal flow of the fluid, an additional pressure drop is present. If the valve is closed by pressure in the containment vessel, only a rather low force is available for closure. In any case, a design using check valves should meet the reliability requirements of testing and effects of single failures.

Certain features of the containment safety systems for eight power reactors in the United States are described in Table 9.3 (ref. 38). This table begins with a listing of the containment valves. In the different plants these valves are grouped in different ways; the most obvious dichotomy would have been to list those valves whose failure to close could lead to external contamination separately from those whose failure could not lead to external contamination, but this is not always followed. In the table, manual and automatic both refer to a remotely operable valve; local refers to a valve that can be closed by a handwheel of other manual operator at the valve itself. The enclosure ventilation is considered separately, since it is such a direct path for potential contamination leakage. It should be noted that the Yankee, Indian Point, and NS Savannah plants normally operate with their ventilation systems sealed.

Table 9.3 also describes the action of the containment valves upon loss of electric power and instrument air. These descriptions are followed by a listing of the signals, or parameters, that must be sensed in order to initiate closure. The number of sensors per parameter and the

Table 9.3. Features of Some Reactor Containment Systems

| Reactor | Number and Type of Valves per Line[a] | | | Action of Automatic Valves on Loss of Power | | Parameters Sensed to Close Automatic Valves and Number of Sensors Per Parameter | System Logic for Automatic Valves | Monitors in Control Room | | Possible Emergency Action if Valve Does not Close |
|---|---|---|---|---|---|---|---|---|---|---|
| | Vacuum Relief | Enclosure Ventilation | Process Lines[b] (Air, Steam, Water, etc.) | Loss of Electric Power | Loss of Instrument Air | | | Signals | Valves | |
| VBWR | None | 1 automatic, 1 local, normally open | 1 automatic and 1 local in lines constituting possible routes for external contamination<br>1 local in other process lines | All valves close | Enclosure ventilation valves close | Varies with valve, but includes 2 to 10 of following:<br>High steam radiation, 1<br>Loss of power, 1<br>High stack radiation, 1<br>High condenser pressure, 1<br>Low reactor water level, 1<br>Low circulating water pressure, 1<br>High steam flow, 1<br>High enclosure pressure, 1<br>Seismic disturbance, 1<br>High enclosure radiation, 1<br>Manual operation possible | Any 1 signal to close valves | Individual lights | Individual lights | Close local valves, handles outside enclosure (also can be closed manually from control room) |
| PWR | | 2 automatic, normally open | None in hydraulic valve lines<br>1 manual in main steam and feedwater lines<br>1 manual or local and 1 check valve in other inlet lines<br>1 manual or local in other outlet lines | Enclosure ventilation valves close (Note: cannot close enclosure ventilation on loss of hydraulic pressure) | Does not apply | High enclosure pressure, 2<br>High stack activity, 1<br>High enclosure air activity, 1<br>Manual operation possible | Any 1 signal to close enclosure ventilation valves only | High enclosure air activity only | Individual lights, enclosure ventilation only<br>1 annunciator | Close enclosure ventilation valves locally; would take considerable time |
| Dresden | 2 enclosure ventilation valves open automatically | 2 automatic, normally open | 1 manual and 1 check in inlet lines<br>1 automatic and 1 check in outlet lines | Motor-operated valves switch to station battery<br>Other valves close | Valves close | Enclosure ventilation valves, all scram parameters, 4 or 6<br>Other valves:<br>High enclosure pressure, 4<br>Low reactor water level, 4<br>Manual operation possible | 2-out-of-4 or 2-out-of-6 to close enclosure ventilation valves<br>2-out-of-4 for other valves | Annunciates only upon high radiation level or steam leak in enclosure | Individual lights and annunciator only on enclosure ventilation valves | Close manually from control room and, in some cases, locally also |
| N.S. Savannah | None | 1 manual, normally closed | 1 check valve in inlet lines<br>1 automatic in outlet lines; most lines have additional local | Close | Close | High enclosure pressure, 3 | 2-out-of-3 to close valves | None | Individual annunciators | Close additional local valve where it exists |
| Piqua | 1 | 2 automatic, normally open | 1 automatic and 1 local in organic coolant lines<br>1 manual and 1 local in feedwater lines<br>1 automatic and 1 local in waste lines<br>2 local (1 inside, 1 outside enclosure) in other lines | Close | Close | All scram parameters, 3<br>High stack activity, 3 | 2-out-of-3<br>Any 1 to close valves | Individual lights and annunciators | Individual lights for enclosure ventilation valves; only temperature, pressure, etc. indicators for other valves | Close locally (would take considerable time) |
| Enrico Fermi | 2 | 1 automatic, normally open, except none in closed-circuit nitrogen cooling loops | 1 manual, locked closed, in purge line; other lines have 1 automatic normally open valve (Note: no water or steam lines in enclosure) | Close | Close | High particulate activity in enclosure, 4<br>High enclosure pressure, 3 | Any 1 of four sensors to close all valves; three additional sensors close certain selected valves | Group annunciators | Individual lights | Close manually from remote station; some also close locally |
| Yankee | None | 1 manual, normally closed | 1 automatic and 1 manual in main steam lines<br>2 check valves in other inlet lines<br>1 trip valve in other outlet lines | No electrically operated valves | Close | High enclosure pressure, 2 | Any 1 to close | 1 annunciator | None | Close locally |
| Indian Point | 2 | 2 automatic, normally closed, in main system; 2 automatic, normally closed, in auxiliary system | 2 automatic in lines constituting possible routes for external contamination<br>2 manual in vital service lines<br>1 automatic and 1 manual in other lines | Enclosure ventilation valves and valves in lines constituting possible routes for external contamination close | Close in lines constituting possible routes for external contamination | High enclosure pressure, 8<br>High boiler water radioactivity, 4<br>Manual operation possible<br>High stack radioactivity, 1<br>Low pressure level, 2 | 4-out-of-8 to close<br>Each sensor closes valves isolating its monitored boiler<br>Closes only enclosure ventilation valves<br>2-out-of-2 to close all valves | Individual lights and annunciators | Individual lights | Close locally; valve handles are in shielded room |

[a] Automatic indicates a valve closed by instruments. Manual denotes a valve operated remotely by an operator in the control room. Local means a valve operated by hand at or near the valve.

[b] In this tabulation it has not been possible to take into account the pressure rating or special conditions that may apply to the systems to which these lines are connected.

system logic, taken together, permit evaluation of the effects of component failure on system operation in terms of safety action when needed and of unnecessary shutdowns.

The tabulations of signal and valve monitors and possible emergency valve action in Table 9.3 pertain to the possible situations in which a human operator might be in a position to take action. In order for him to act, he must have the knowledge that action is needed and the means to effect the action.

From a study of the table, it is evident that some containment systems are not designed to have the same reliability as neutron safety systems. Although three independent instrument channels are considered standard for neutron flux, or reactor power, many containment systems rely on fewer than three. The techniques of redundancy and coincidence are used in only one-half of the systems listed. In almost all cases a single electrical or pneumatic signal is sent to all valves; failure of the signaling device or of its energy source results either in automatic closing of all valves or in inability of any of the valves to close. Such a situation in a neutron safety system is now widely recognized to be poor practice. A large body of experience with false scrams has testified to the need for more sophisticated techniques to ensure that such a system will act when it is required and that it will not act spuriously. Since correct action by the containment system is essential to protect off-site personnel (the general public) in the event of a nuclear incident, it is incumbent upon the nuclear power industry to examine critically the criteria on which the design of such systems is based. It would be intolerable for the containment to be ineffective because of instrument or control failure.

### 9.8.2.2 Air Locks

The air-lock doors and the associated interlock to prevent both doors being open simultaneously are in the category of a protective system. All the considerations of reliability should apply, including testing, independence of channels, and effect of a single failure, whether the interlock is a simple mechanical device or a complex electromechanical system.

A minimum of two independent channels should be provided to ensure that one door is closed and locked before the other door is opened. The human operator, exercising administrative control, should not be considered as one of these channels, since he may become accustomed to the interlock preventing his making errors and so become dependent on the interlock. Diversity may be employed to good advantage. If the operation of the interlock depends upon devices that detect position of a door, these devices should indicate that the door is securely locked as well as closed, especially if there may be any pressure differential tending to open the door. The loss of electric power or pneumatic or hydraulic pressure should not allow the interlock system to fail to danger and thus allow both doors to be open simultaneously.

### 9.8.2.3 Spray Systems

Spray systems that are necessary in order for containment to be achieved or maintained must be classed as protective systems. A water spray may be needed to prevent the containment vessel from reaching a temperature above its design limit or to prevent the internal pressure from rising above its design limit.

As in other protective systems, a minimum of two independent channels to obtain the cooling spray is required, and the system must provide protection despite the failure of a single portion of the system. A channel of this system includes the sensor, amplifiers, relays, connections, control valve, and associated spray nozzle, piping, water valves, and water supply. If power-driven pumps are required, then such pumps, their prime movers, and any electric power supplied to run the pump motors are also part of the spray channel. In those systems requiring a cooling spray immediately after the incident needing containment, water supplied from elevated tanks would eliminate the absolute necessity of a continuous assurance that a pump was operable and that power was available for starting and running it. Two independent water supplies, each separately having sufficient capacity for the cooling required of the system, should be provided. Wherever it is necessary to secure water directly from a single large body, such as a lake or river, for both channels, then an independent and separate method of obtaining the water from the source should be employed for each channel.

As in other protective systems, the reliability requirements should be satisfied. In particular, operability of the spray system should be assured through adequate monitoring and testing.

### 9.8.2.4 Other Systems and Related Components

In addition to valves, air locks, and spray systems, other systems and their associated components may be vital to achieving containment. For example, systems using blowers, filters, and a stack may be employed to limit the spread of radioactive gases, or systems for the removal of afterheat from the reactor may be necessary if containment is to be maintained. All such systems must have the high reliability required of protective systems for containment. The sources of energy and the heat sinks required by these systems constitute important components whose reliability must be assured. The requirements for reliability given in Section 9.8.1 should be satisfied. The important considerations are redundancy, using separate and independent channels, and adequate testing or monitoring to determine that the system is operable.

## 9.9 ELECTRIC POWER SYSTEMS

The economic incentive for uninterrupted production of power in a large nuclear plant and the need for plant safety demand a careful arrangement of the electrical system interconnections and distribution.[53] The power system is important both for the distribution of the power produced and for the continued supply of power to essential equipment within the facility. It should be capable of furnishing these services despite a failure of the power system in locations internal or external to the plant. The requirements of these functions establish a demand for reliability of the electrical connections to, and the distribution system within, a nuclear power plant that is seldom matched in industry.[54] The continuity of power service to the consumer cannot always be ensured, but the power system design must ensure a safe shutdown of the plant. Power must be available at all times to vital systems and equipment, such as the shutdown and emergency cooling system, ventilation equipment, and filtering systems, which are provided either to prevent the occurrence of serious accidents or to minimize the consequences thereof. The need for dependable sources of power for control systems and protective systems is obvious, since the loss of this power would necessitate costly shutdowns, if not result in serious accidents. Thus, in addition to one or more external power lines, most nuclear plants must include one or more internal sources of emergency electrical power, such as diesel-driven generators and batteries.

Although dependability of the power sources is a requirement common to all types of nuclear power plants, the type and complexity of the electric power systems may vary to a great extent from one facility to another. In particular, the urgency of the need for emergency cooling influences the electric system designs of the various plants. The principal features of the electric systems of 20 power reactors are presented in Table 9.4 as illustrative examples.[54] As indicated by the headings in this table, electric power for nuclear plants may be placed in the major categories of primary, auxiliary, and emergency power. An additional, but vital, application is in the area of electric power for the instruments of protective systems and control systems listed under reactor control power and reactor control emergency power in the table.

### 9.9.1 Primary Power System

The primary power system includes the external power lines, the primary bus arrangement within the plant, associated transformers, and the plant generators. This system provides the source of power for the auxiliary power system, as well as the means by which power is transmitted from the plant. The availability of primary power, then, is principally of economic concern, although the requirements for auxiliary power will affect the design of the primary power system.

In many instances, the site of the plant will have a considerable influence on the extent to which the designer can go in obtaining multiple external power sources. In the case of large central stations, there is an economic incentive for delivering uninterrupted power to the consumer, and therefore at least two connections to the electrical power grid are desirable in order that the loss of one connection will not require a shutdown or a decrease in the level of power output. These power lines to the grid should run along divergent rights-of-way to minimize the frequency of simultaneous interruptions during severe storms.

Table 9.4. Electric Power Systems of Various Nuclear Power Plants[a]

| Reactor | External Primary Power | Line Rights-of-Way | Primary Bus Arrangement | Plant Power Generation (Mw) | Number of Generators | Plant Auxiliary Power Distribution Feeders |
|---|---|---|---|---|---|---|
| PWR | Two 138-kv lines | Separate, closed loop | Solid | 68 gross<br>60 net<br>[100 gross] | 1 | Four 2400 volts; four 480 volts for reactor; four 480 volts for turbine-generator; two 480 volts for screen house |
| Indian Point | Three 138-kv lines | Common, radial | Sectionalized, normally closed automatic tie breaker | 275 gross<br>255 net | 1 | Four 13.8 kv and multiple 440 volts |
| Yankee | Two 115-kv lines | Separate, closed loops | Solid | 145 gross<br>136 net | 1 | Three 2400 volts, three 440 volts |
| NS Savannah | None; plant primary power supplied by 1500-kv turbine-generators | | Sectionalized, normally closed automatic tie breaker | | | |
| Saxton | [Four 115-kv lines] | [Three separate] | | [5.75 gross]<br>[5.0 net] | [Existing] | |
| SPWR | | | | 16.5 net | 1 | |
| VBWR | | | | 5 | 1 | |
| Dresden | Five 138-kv lines | Separate, closed loops | Sectionalized, normally closed tie breaker | 192 gross<br>180 net | 1 | Two 4.1 kv, four 480 volts |
| Elk River | [Four 69-kv lines] | [Separate] | Solid | 22 | 1 | Two 440 volts (one standby) |
| Pathfinder | Two 115-kv lines | Separate, closed loop | Solid | 66 gross<br>62 net | 1 | Two 2.4 kv |
| Humboldt Bay | External lines not indicated; two existing conventional units on common bus | | . | 60 gross | 1 | Two 2.4 kv (one standby) |
| BONUS | Two 38-kv lines | | Solid | 17.3 | 1 | One 480 volts |
| Big Rock Point | | | Solid | 75 gross | 1 | Two 2.4 kv (one standby) |
| Hallam | | | | 75 gross | 1 | Two 2.4 kv (one standby) |
| Enrico Fermi | Two 120-kv lines | | Sectionalized | 104 gross<br>94 net | 1 | Three 4.8 kv (one standby) |
| EGCR | One 13.8-kv line | Radial | Solid | 25 | 1 | Two 2.4 kv, four 440 volts |
| FWCNP | Two 65-kv lines | | Solid | 50 | 1 | Two 2.4 kv (one standby), two 600 volts |
| Peach Bottom | One 220-kv line<br>[Reserve 33-kv line] | Separate, radial | Solid<br>Generator ACB | 40 | 1 | Two 2.4 kv; two 480 volts |
| Piqua | Two 4.1-kv lines | Common | | 11.4 | Existing | |
| CVTR | Four 114-kv lines; one 66-kv line | Separate, closed loop | Two solid | 30 | Existing | Two feeders from existing steam plant |

[a]Entries in brackets represent changes made by the reactor designer since the publication of the hazards reports used for this comparison. These changes are only listed for additional information.

Table 9.4 (continued)

| Reactor | Auxiliary Bus Arrangement | Plant Emergency Source | Reactor Control Power | Reactor Control Emergency Power |
|---|---|---|---|---|
| PWR | Four sections for reactor, normally open tie breaker; four sections for steam generators, normally open tie breaker | One 500-kw diesel, manual; one 23-kv line | 120-volt a-c<br>125-volt d-c (safety system) | 125-volt battery, manual transfer |
| Indian Point | Four sections, normally open tie breaker | None except as derived from external primary power lines | 120-volt a-c alternators | |
| Yankee | Three sections, normally open tie breaker | One diesel, manual | 120-volt a-c alternators | One diesel |
| NS Savannah | Two sections, normally closed tie breaker | Two 750-kw diesels, automatic; one 300-kw diesel, automatic start, manual switching | Two 120-volt a-c alternators (one standby)<br>One 120-volt, a-c from auxiliary transformer | 120-volt a-c from auxiliary transformer |
| Saxton | | [Existing steam plant with two turbine-generators] | | [120-volt a-c from station emergency source] |
| SPWR | | | | |
| VBWR | | | | |
| Dresden | Two sections, normally open tie breaker | One 500-kw diesel, automatic; one 34.4-kv line | 125-volt d-c battery; two 120-volt a-c alternators (safety system) | |
| Elk River | Solid | One 35-kw diesel, automatic | 120-volt a-c alternator; d-c for safety system | 120-volt a-c from station source |
| Pathfinder | [Two sections, each with one normal and one emergency 2.4-kv feeder] | [One 125-kv diesel] | [One a-c alternator driven by battery-power motor] | |
| Humboldt Bay | Solid | [Bus from existing units; one diesel] | Two 120-volt a-c alternators | |
| BONUS | Solid | One 100-kw diesel | 120-volt a-c | 120-volt a-c standby feeder |
| Big Rock Point | Solid | | Two 120-volt a-c alternators with gasoline engine as emergency drive | |
| Hallam | Solid | One 400-kw diesel, automatic | Two 120-volt a-c alternators | 120-volt a-c station feeder |
| Enrico Fermi | Solid | One 80-kva diesel | 120-volt a-c alternator; 125-volt d-c | |
| EGCR | Two sections, normally closed tie breaker for 2.4-kv feeders; two 2-section buses, normally open tie breaker for 440-volt feeders | Two 1000-kw diesels, automatic | Two 120-volt a-c alternators; two batteries | |
| FWCNP | Two sections for 2.4-kv feeders; two sections, normally open tie breakers for 600-volt feeders | One 200-kw diesel | Two 250-volt batteries | |
| Peach Bottom | Two sections for 2.4 kv, each with one normal and one emergency feeder | Reserve 33-kv line; one 1000-kw diesel, automatic; coastdown power on main turbine-generator set inertia | 120-volt a-c | Two 3-unit motor-generator sets; one transformer from station emergency source |
| Piqua | | One 250-kw diesel, automatic | One 120-volt a-c alternator; two batteries on same bus | 120-volt a-c station feeder |
| CVTR | Two sections, normally open tie breaker | One feeder from hydro plant | | |

### 9.9.2 Auxiliary Power System

One of the functions of an auxiliary power system is to distribute power to equipment such as the pumps and blowers necessary to operate the plant normally. Auxiliary power systems also distribute power to electrically powered components of protective systems in the plant. Examples of the components necessary for protection are blowers associated with filters for achieving containment and pumps in emergency afterheat removal systems necessary to maintain containment.

For normal operation of the plant, a minimum of two feeder circuits from the primary power system to the auxiliary system should be provided. In the auxiliary power distribution system, the connection of these feeders to either a solid bus or to sectionalized buses operating as a solid bus (with the tie breaker normally closed) will allow for the loss of one source without recourse to automatic transfers or reduction in power level. The presence of a tie breaker permits isolation of a bus fault; however, the reliability of a properly designed solid bus is probably as high as or higher than the reliability of automatic switching equipment. Another, and potentially better, arrangement utilizes two or more bus sections with normally open tie breakers. In this case, load diversification can assure at least a partial load operating in the event of failure of one of the multiple sources until full-source supply can be restored by manual or automatic transfers to the operable sources. With large reactors, a satisfactory auxiliary system utilizes multiple centers with broad load diversification and interconnecting facilities for the many bus sections. A temporary partial loss of load is unavoidable with such arrangements in the event of loss of an auxiliary feeder. Directly coupled rotary sets of the synchronous motor-flywheel-alternator type may sometimes be used to cover the transient period between line transfer or even during the starting of a diesel-driven emergency generator. In addition, this drawback can be avoided through the installation of dual full-capacity components, such as circulating pumps, that operate in parallel, with each pump connected to a different source of power. This arrangement has great merit, even though the capital investment is increased.[54]

These portions of an auxiliary power system that are required to distribute power to plant protective systems must meet the reliability requirements of protective systems. The power distribution circuits involved in the protective system should have multiple and separate channels over which the power may be transmitted. Each circuit should deliver power to a separate piece of equipment, such as separate motors, in order to maintain independence of the channels. Dual, full-capacity pumps or blowers should be provided.

### 9.9.3 Emergency Power Sources

The reliability of the auxiliary power system can be no greater than the reliability of the sources of power provided. As previously discussed, power is required to operate protective systems in certain instances.

Since the reliability required of protective systems supplied from the auxiliary distribution system must be much higher than the reliability of most primary power systems, an emergency power supply is necessary.

The station generator should not be considered to have the reliability required for protective system service. There can be no positive assurance that the generator will continue to operate after a complete severance from the outside system. Furthermore, after a reactor shutdown, the station generator is not available at all as a source. The problem of emergency power is simplified in cases where the nuclear plant is integrated with existing steam and hydroelectric plants at the same site. A similar advantage will be obtained when nuclear plants accommodate two or more reactors. Some reactors allow a considerable lapse of time upon complete power loss before damage to certain components can be expected. Even so, an independent emergency source in the plant is the best safeguard against the most extreme eventualities.[54]

In the majority of the plants, a complete failure of the primary supply is offset by providing one or more diesel-driven generators that are automatically started and tied to the essential postscram loads. Other plants rely on the availability of an emergency line from a different system before resorting to the use of the diesel-driven generator. Failure of a diesel engine to start or a delay in its operation could be serious.[55] Charger-battery-inverter sets are sometimes employed when the load is relatively small, but such sets are less desirable than diesel-driven generators in the case of large motor loads. Frick has proposed to make use of the large coastdown inertia of the main turbine generator by utilizing an auxiliary generator on the same shaft with the main unit.[56] The auxiliary generator would supply the emergency power needed during the coastdown until the diesel or some other emergency power source could take over the duty.

### 9.9.4 Power for Instrumentation

Much of the instrumentation used in control systems and protective systems has been designed to operate on 120-v a-c power. The failure of such power is usually considered to be unacceptable from the standpoint of either serviceability or safety; therefore, reliable sources must be provided. Direct current to alternating current conversion from batteries is quite common.

Coincidence is often employed in protective systems; however, the benefits of coincidence can be fully attained only by the application of coincidence throughout these systems.[49] Independent and separate power supplies should be provided for each instrument channel of a coincident and redundant group.

Protective channels sometimes employ direct current for instruments. Batteries are inherently more reliable than rotary equipment, and the use of solid-state devices operating on direct-current sources would allow alternating-current sources to be abandoned in favor of batteries. This simplification of source requirements should result in greater reliability of the safety systems.[54]

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# 10. PERFORMANCE TESTS

R. F. Griffin*     G. H. Dyer*

## 10.1 INTRODUCTION

A containment structure must be thoroughly tested to ensure both its structural integrity and its leaktightness. Structural integrity or strength tests of containment vessels are generally conducted in accordance with established code requirements and are similar to the strength tests of other large pressure vessels. On the other hand, leakage tests of containment vessels require special methods and special care because of the very low maximum allowable leakage rates normally specified. A distinction should be noted between two chronological phases of leak testing. The first phase relates to leakage tests usually performed by a containment vessel fabricator, generally on the bare shell, to demonstrate compliance with the customer's acceptance leakage rate, whereas the second phase relates to tests performed by the operator on the completed containment structure both prior to operation and at subsequent intervals thereafter to demonstrate continued compliance with a leakage rate deemed acceptable from the safety standpoint.

This chapter describes tests performed on containment systems to demonstrate their ability to control the spread of radioactivity in the event of an accident in the nuclear system. Obviously it is seldom possible to test a containment system at the specific conditions of the postulated maximum credible accident. For example, a high temperature may be associated with the postulated accident that would be impractical or impossible to simulate for the leakage-rate test. Notwithstanding, the contaiment design must take into consideration the effects of temperature and thermal gradients upon the production or alteration of leakage paths, and the leakage-rate tests must reflect conservatism in the extrapolation of leakage rates at test conditions to the anticipated values at accident conditions.

The strength and leakage criteria and the requirements for testing presented here are based on applicable codes and standards and on engineering practices followed for most of the existing U.S. nuclear plant containment vessels.** These criteria are based on conservative assumptions as viewed by existing understanding of containment technology and are subject to change as additional experience and experimental evidence is accumulated.

Included in this chapter are discussions of pressure testing to indicate structural integrity, local leak testing to detect and locate sources of leakage, integrated leakage-rate testing to confirm leaktightness, and the retesting that may be required to reconfirm leaktightness. Tests of individual containment components before they are assembled into the containment structure and weld inspections, such as radiography, dye

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*Bechtel Corporation, San Francisco, California.

**The analysis of leakage determinations, presented in Secs. 10.4.5 and 10.4.6, was adapted from R. J. Maccary et al., Leakage Characteristics of Steel Containment Vessels and the Analysis of Leakage Rate Determinations, USAEC Report TID-20583, May 1964.

penetrant, or magnetic particle tests, are not discussed, since these are considered part of construction details and are discussed in Chapters 8 and 9.

Primary emphasis is placed on the testing of welded steel-shell containment vessels, reflecting the great predominance of this type of containment in current use for U.S. power reactors. The much greater experience in testing containment structures of this type is reflected in the availability of standards for this purpose. However, this chapter also deals briefly with other types of containment, such as conventional buildings, concrete structures, pressure-suppression systems, and multiple-barrier systems, for which new testing techniques or special adaptation of the present techniques may be required. Industrial practice directly applicable to the testing of the unique features of these containment concepts is not yet well defined, so the information contained in this chapter is necessarily somewhat speculative as to the methods that may be needed to test these systems adequately.

Two standards outline present industrial practice for testing containment structures. They are

1. Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors (see Appendix E; specifically, Section 15 — Pressure Testing for Strength, Section 16 — Leakage Testing, and Section 17 — Periodic Inspection and Testing), prepared by Subcommittee N6.2 of Sectional Committee N6 of the American Standards Association (approved by ASA, April 1965),
2. Proposed Standard for Leakage Rate Testing of Containment Structures for Nuclear Reactors, prepared by Subcommittee ANS-7 of the American Nuclear Society Standards Committee.[1]

The criteria presented in these standards are considered to be representative of current practice and are referred to throughout this chapter. As is typical of standards, they do not represent the best practice but rather the *minimum* acceptable requirements. Reference should be made to Chapter 2 for a fuller discussion of existing and proposed standards and codes. Since containment technology is, metaphorically speaking, in its minority, it is to be anticipated that these standards will undergo considerable revision as the art is advanced.

## 10.2 STRENGTH TESTING

### 10.2.1 General

Because of the possibly disastrous consequences of their failure during an accident and because of the relative ease with which they can be structurally tested, pressure vessels are customarily tested to greater than design pressure before being placed into normal operation. Although differing in many respects from typical pressure vessels, most reactor containment structures built to date have been made of steel and designed and tested in accordance with accepted pressure vessel codes. The requirements and procedures for testing these "conventional" containment vessels are quite well defined and are discussed in Section 10.2.2 of this chapter.

The strength-testing procedures for other types of containment structures are not defined by code requirements and usually are established for each structure on a case basis. Special considerations for testing some of these other types of structures are discussed briefly in Section 10.2.3. The present scope of practice for both conventional and nonconventional containment has limited strength testing to the structure's capability to resist internal pressure loading. Other loading conditions, for example, pipe reactions, air-lock loads, accident jet reactions, and accident missiles may provide sufficient stressing in some instances to require verification other than that afforded by analysis.

## 10.2.2 Steel Containment Vessels

### 10.2.2.1 Code Requirements

Most steel containment vessels have been designed and tested in accordance with the ASME Code for Unfired Pressure Vessels (UPV Code),[2] and a number of Nuclear Case Interpretations have been issued by the ASME Boiler and Pressure Vessel Committee to clarify the application of this code to nuclear vessels. Case 1272N-5, which was the principal interpretation defining the requirements for containment vessels, has now been incorporated into Section III, which is applicable to a design pressure greater than 5 psig. (Refer to Chapter 2 for a discussion of the legal aspects of codes.) The ASA Standard for steel containment structures (Appendix E) is also applicable to containment vessels with design pressures above 5 psig and, with some modification, even to vessels with design pressures below 5 psig. At least one low-pressure containment vessel has also been designed in accordance with API Standard 620,[3] which applies to vessels with design pressures up to 15 psig and operating temperatures up to 200°F.

The new section, Section III, of the ASME Boiler and Pressure Vessel Code was published specifically to cover vessels used in nuclear installations (see Appendixes C and D).[4] Section III classifies containment vessels as Class B vessels, and it applies to such vessels having a design pressure greater than 5 psig. Subsection B covers Class B vessels and incorporates the provisions of the UPV Code and the latest Code Case Interpretations for containment vessels.

The UPV Code requires that vessels designed in accordance with its provisions be pressure tested either hydrostatically to 1.5 times the vessel design pressure or pneumatically to 1.25 times the vessel design pressure. API 620 also requires a pneumatic pressure test of the completed vessel to 1.25 times the vessel design pressure. Consequently, many containment vessels have been pneumatically tested to 1.25 times design pressure. However, Section III specifies that pressure tests for containment vessels be conducted at not less than 1.35 times design pressure when hydrostatic tests are made and at 1.15 times design pressure when pneumatic tests are made. This reduced requirement comes about because Section III allows design membrane stresses for containment vessels to be 1.1 times those allowed for other Code-designed pressure vessels in lieu of the 10% increase in pressure permitted for vessels fitted with pressure protective devices. The pressure test requirements of the various codes are summarized in Table 10.1.

For most pressure vessels, hydrostatic pressure tests are preferred to pneumatic tests because the high energy content of compressed air within a vessel makes pneumatic tests more hazardous. A hydrostatic pressure test is generally used when the test pressure is large compared with the hydrostatic head of the water required to pressurize the vessel. Consequently, for relatively small higher pressure vessels than are usually used for containment vessels, hydrostatic tests are employed. Since containment vessels are not usually designed to hold liquid, the pneumatic pressure test is almost always used. The vessel and its supports normally are not capable of supporting the mass of water required for the hydrostatic test. Also, it is often convenient to use the pneumatic pressure test for containment vessels because this test is usually followed immediately by the initial vessel leakage-rate test, for which air is used.

In only a few special cases, for smaller vessels, has a containment vessel been hydrostatically pressure tested. For example, the NS Savannah containment vessel, which has a gross volume of only about 40,000 $ft^3$, was hydrostatically pressure tested at 173 psig.[5] Also, the containment vessel now being used for the Molten Salt Reactor Experiment (MSRE) at the Oak Ridge National Laboratory was originally tested hydrostatically to a pressure of 200 psig. However, most containment vessels have been pneumatically tested in accordance with the UPV and API 620 Codes. Accordingly, only the pneumatic pressure test will be considered further in this chapter.

Table 10.1. Pressure Vessel Test Pressure Requirements

| Code | Ratio of Test Pressure to Design Pressure | |
|---|---|---|
| | Hydrostatic Test | Pneumatic Test |
| UPV Code, Section VIII | 1.5 | 1.25 |
| Code Case 1272N-5 | 1.35 | 1.15 |
| UPV Code, Section III | 1.35 | 1.15 |
| API Code 620 | 1.25 | 1.25[a] |

[a]Actually, a combination hydrostatic-pneumatic test.

#### 10.2.2.2 Test Procedure

The pneumatic pressure test of a containment vessel is usually conducted before any concrete or equipment has been installed within the structure, unless, of course, a multistage (see Sec. 8.4.2) construction scheme is used. Air locks and doors that are part of the pressure-containing structure normally are installed and subjected to the pressure test. Both air-lock doors should be pressure tested; this may be done by pressurizing the air lock after the containment vessel test pressure has been reached with the inner door closed. Smaller penetrations that are to be used for piping and wiring may be made in the vessel prior to the test, but they are blanked off during the pressure test. Since the function of the test at this point is the confirmation of strength integrity, subsequent local and integrated leak-detection and leakage-rate tests are performed on these smaller penetrations after installation of the equipment, as discussed in Sections 10.3 and 10.4. Individual strength tests of the completed penetrations for piping, wiring, valves, and bellows may be done individually for each unit or later as a local test.

Portable compressors are used to pump outside air into the vessel. When the pressure in the vessel reaches some nominal pressure, a small fraction of the design pressure, pressurization is stopped to permit inspection of the vessel for visible signs of distortion or gross leakage. A soap bubble test of all weld seams and closures on the entire external surface is usually conducted at this time to detect any leaks that might affect the integrity of the vessel or prevent it from attaining the test pressure. If leaks in the vessel proper, not the temporary test equipment, are discovered, the vessel is depressurized and the leaks are repaired before proceeding with the pressure test. Leaks in the temporary test equipment may be sealed without depressurization. Repairs requiring release of the low pressure are normally inspected again at the same pressure. Pressurization is then resumed, and the vessel pressure is increased in increments until the test pressure is reached. Paragraph UG-100 (c) of the ASME UPV Code, Section VIII, specifies steps of approximately one-tenth of the test pressure. Although not a current general practice, consideration should be given to intermediate leak tests at each pressure plateau. Such tests could provide a rational basis for interpretation of subsequent tests. When the test pressure has been reached, it is held for approximately 1 hr to complete the proof-of-strength test. Following the proof test, Paragraph UG-100 (c) of Section VIII specifies inspection at four-fifths of the test pressure; under Section III this inspection could conveniently be accomplished at design pressure. Since the full envelope of the containment system is incomplete at this point, without electrical penetrations, isolation valves, etc., the strength test is not necessarily complete in its fullest sense. Operational experience, previous test data, individual tests, or other data should be requisites for establishing the strength integrity of those parts that could also be subjected to accident conditions. If no new leaks or other difficulties have been encountered, it is then often convenient to proceed with leak-detection and leakage-rate tests. For this purpose a second soap bubble test is sometimes conducted, and the integrated leakage-rate test of the vessel is performed, as described in Section 10.4. Section 10.5 presents a step-by-step procedure of a typical initial integrity-testing sequence, including both pressure testing and leakage testing.

After the pressure test has been completed, all seams of the vessel should be visually inspected for any signs of damage due to the test.

This may be accomplished during the soap bubble test immediately following the strength test. This procedure requires that all seams of the vessel be accessible at the time of the test and precludes placing any concrete either inside or immediately outside of the vessel walls prior to the test. However, UPV Code Case 1272N-5 and Section III allow an exception to this procedure for multiple-stage construction, in which concrete is placed over some of the welded joints before the vessel is completed, provided that all joints are completely radiographed and that there are no penetrations in the area covered by the concrete. When the vessel is completed, a pneumatic pressure test is conducted, as described above, except that following the test it is no longer possible to inspect all welds.

### 10.2.2.3 Equipment

The equipment required to conduct a pneumatic pressure test includes a compressor or bank of compressors capable of pressurizing the vessel to full test pressure in a reasonable period of time and a reliable pressure gage with a range covering the full range of test pressures. A pressure-relief device may be installed for the pressure test to prevent overpressurizing the vessel in the event of equipment or instrumentation malfunction or a rapid atmospheric temperature increase. In many instances pressure monitoring with two independent accurately calibrated gages has been an adequate procedure. Because of the hazards associated with pneumatic pressure tests, the compressor controls and pressure indication must be placed at a location remote from the vessel. In some cases, strain gages are placed on the vessel to determine actual strains where the stress patterns are complex and design calculations cannot establish the stresses conclusively. A general practice has been to employ the test equipment only for the duration of the initial test. In some instances it may prove economical to incorporate the devices in the containment complex in order to reduce down time required for subsequent leak tests. For such instances, provision must obviously be made to isolate the equipment, for example, the relief device, in order not to violate the containment during normal operation.

### 10.2.2.4 Precautions

The hazards associated with a pneumatic pressure test are greater than those of a hydrostatic pressure test because of the considerable amount of stored energy contained by a gas under pressure. Accordingly, special safety precautions are required during a pneumatic pressure test. The principal requirement is that all personnel maintain safe distances from the vessel while it is being pressurized. Such safe distances will depend upon the factors involved that determine the missile characteristics possible and upon the availability of missile protection, either natural or specially provided. Normally greater distances are specified for personnel not required to assist with the test than for test personnel. Access to the exterior surface of the vessel to read instruments and to check for leakage or signs of distress is permitted only when the pressure is being held at a constant value and then only by qualified personnel specifically assigned to these tasks. At design pressure, access is permitted only after the vessel has successfully passed the pressure test at full test pressure, since a failure at design pressure is then highly unlikely. A person primarily responsible for safety considerations during a test should be present whenever the vessel is pressurized to ensure that all safety precautions are properly enforced.

The vessel must not be pressurized if it is at or near the temperature below which brittle fracture of the vessel steel is possible. A discussion of the properties of steel is presented in Chapter 8.

Extra precautions should be taken to assure that doors and other mechanical closures are adequately fastened. These closures should be double checked before the vessel is pressurized.

#### 10.2.2.5 Scheduling

The pneumatic pressure test of the containment vessel must be closely integrated into the construction schedule, since the test requires a significant length of time and little or no other work can proceed in the vicinity while the test is in progress. The test is usually performed as soon as possible after the vessel has been completed and all openings have been sealed. Pressurization of the vessel may take several days because of the great quantity of air required. If soap bubble leak tests are conducted after the first pressure increment and again after the pressure test has been completed, the total test period could be increased by at least another half day. As stated previously, it is normal practice to follow the pressure test with the initial integrated leakage-rate test to avoid having to pressurize a second time.

Once the vessel has successfully completed the pressure test, additional pressure testing is not normally required unless major modifications are made to the vessel. What constitutes a "major modification" is left to the discretion of the responsible inspector but is usually considered to be a modification that results in a significant change in the stress patterns of the vessel. Cutting a large, temporary construction opening in the side of the vessel and then rewelding usually is not considered to be a major modification. This differs somewhat from the requirements for leakage testing; new openings must be leak tested either locally or with an integrated leakage-rate test and periodic leakage-rate tests as required to provide evidence of compliance with the acceptable leakage rate. Adequate corrosion protection or corrosion allowance must be provided for the vessel, as required by paragraph 1330 of the new Section III, and once its initial structural strength has been proved, it is considered adequate for the design lifetime of the structure. Evidence of continued structural integrity will necessarily demand continuous surveillance, maintenance, and, in some cases, retests.

### 10.2.3 Other Containment Structures

The requirements for strength testing containment structures other than the conventional steel pressure-containing type are not standardized and must be established in each case. Typical examples of likely test requirements are given in the following discussions of various types of containment structures.

### 10.2.3.1 Conventional Buildings

Conventional-type building structures have been used to supplement or to act as primary containment barriers in cases where no significant pressure buildup within the building can be hypothesized. In these cases, normal building codes are used for the structural design, and the structure is not subjected to a pressure test. Leakage-rate tests that may be conducted at small pressure differentials may impose a substantial structural load on the building, but they are not intended as structural tests. Of course, the design of the building must take into consideration the load imposed by the leakage test pressure. Leakage-rate testing of conventional low-leakage buildings is discussed in Section 10.4.13.

### 10.2.3.2 Concrete Containment Structures

Reinforced-concrete structures are normally built in accordance with standard ACI-318 (or ASA A-89.1), Building Code Requirements for Reinforced Concrete. Pressure tests for demonstrating structural integrity are not required under this code, largely because reinforced concrete is not often used for pressure vessels. However, if the design pressure of the concrete structure is sufficiently high that good engineering practice dictates a pressure test, the test requirements of a standard pressure vessel code, such as the UPV Code, can be adapted on a case basis. This was done for the steel-lined concrete suppression chamber of the Humboldt Bay plant (see Sec. 10.2.3.3). A provision for structural testing of concrete structures will probably become a standard requirement of licensing acceptance. Until such time, the spirit of the ASME UPV Code will be applied; concrete vessels are presently being tested accordingly.

Applying internal pressure to a reinforced-concrete structure would cause cracking of the concrete at a pressure well below that at which the design stress of the reinforcing bar was reached. This problem was avoided in a recent experimental structural test of a rectangularly shaped reinforced-concrete containment structure by maintaining the tensile stresses well within the tensile strength of the concrete during the test.[6] This test demonstrated the difficulties of accurately predicting stresses in structures of this type, particularly at locations near reentrant corners and wall openings.

A metal or other liner, such as a bridging epoxy, is used to assure low leakage from a reinforced-concrete containment structure. The liner does not add to the structural strength, but it maintains leaktightness, even if the concrete cracks when the structure is pressurized beyond the tensile strength of the concrete. Prestressed concrete structures are designed to maintain the concrete in compression and thus prevent its cracking. Nevertheless, since concrete is relatively porous, a metal liner probably would be needed in prestressed structures if a high degree of leaktightness were required.

10.2.3.3 Pressure-Suppression Containment

The pressure-suppression system used in the Humboldt Bay plant is a special case of a steel system (the drywell and vent piping) and a steel-lined concrete structure (the suppression chamber). Where possible, the load-bearing steel portions of structures of this type are pressure tested according to the UPV Code and the Code Cases prior to pouring or grouting concrete around them so that all seams are accessible for inspection following the pressure test. Although the design of the drywell and vent piping is based on dynamic loading conditions, these structures are tested statically to greater than the maximum expected dynamic pressure by providing suitable temporary closures on the vent piping. The Humboldt Bay drywell and vent piping were designed as a code vessel and were tested to 1.25 times the design pressure. Although the Humboldt Bay suppression chamber is a concrete structure with the steel liner providing only leak-tightness, a specification of the licensing agreement was that it would be pressure tested to 1.25 times the design pressure to meet the intent of the UPV Code, even though the Code does not apply to structures of this type.

10.2.3.4 Multiple-Barrier Containment

The primary structural member of a multiple barrier containment system may be any one, or all, of the individual barriers. As such, all barriers internal to the outermost structural barrier must transmit any remaining internal pressure load to this outer structure, and any structural test must demonstrate the ability of all internal barriers to transmit this load, as well as the ability of all structural members to withstand their respective shares of the imposed load.

In the NS Savannah and Indian Point containment systems, the procedures for testing steel vessels applied to the inner shell only because the primary pressure load is borne by the internal steel shell and the benefit of release attenuation from using the surrounding structures as additional containment barriers was achieved without significant modifications to the initial design. In potential future types of multiple-barrier containment structures, such as the type that was proposed for the Ravenswood plant in New York[7] or such as may be proposed in conjunction with pressure-suppression, underground, or other combination systems, it is possible that a standard requiring a static structural proof test will evolve. However, no such standard exists to date. If such a standard is prepared, it is likely to include provisions for demonstrating the validity of both load-transmission characteristics and structural characteristics, as well as for subsequent examination of the system for damage.

## 10.3 LOCAL LEAK TESTING TECHNIQUES

### 10.3.1 General

Local leak tests are performed to detect and locate leaks in the containment vessel shell or containment components so that they may be repaired prior to the integrated leakage-rate test of the entire containment vessel. These local tests, although, for the most part, very sensitive, have in the past been principally qualitative. Usually, no attempt has been made to measure the rate of leakage out of the leaks detected, and since the tests are usually performed over a limited area, there is not positive assurance that all leaks have been detected. Since it is generally presumed that penetrations are the most flagrant leak producers, attention has recently been devoted to the possibility of utilizing local leak tests or grouped local leak tests as means for continuous monitoring of the leaktightness of the containment. If the characteristics of the local leakage rates were established as a function of pressure in conjunction with the integrated leak test, a useful correlation might be developed to permit such continuous monitoring.

Local leak tests can be performed on various containment components before they are installed into the vessel or on individual components after the vessel has been completed. In the latter case, local tests may be conducted in conjunction with the containment vessel pressure test. Local leak tests also may be required at containment alterations, such as new penetrations, and to detect and locate leaks that develop after the vessel has successfully completed an integrated leakage-rate test. The use of vacuum boxes or locally pressurizable bulkheads obviously implies that the geometry of the leaks is not sensitive to the stress condition of the containment shell, since this method cannot approach the pressure differentials that would be felt by the containment vessel in the event of an accident. In the employment of such devices, there must be reasonable assurance that such an effect does not exist. In some cases it may be possible to use some of the local leak testing techniques described in this section to perform integrated leakage-rate tests. In these cases, a measurement of leakage rate would be included in the leak test. Section 10.4.10 discusses how some of these methods normally used for local leak testing may be refined to give an accurate indication of the integrated leakage rate of the entire containment vessel.

The ability to use any particular leak testing technique in a quantitative fashion depends to a large extent on the ability of the field activity to carefully control the conditions under which the test is being run. Thus, when a minimum detection limit of less than 0.01 $ft^3$ per day is specified (see Table 10.2) for air bubbles rising through a water blanket, it should be realized that this sensitivity is dependent upon observing 1/16-in.-diam bubbles being released at a rate of one per second. Observation of such bubbles in a relatively small, well-lighted test area should be reasonably possible, but this sort of test phenomenon might easily be overlooked under other conditions.

Table 10.2 indicates the approximate sensitivity of various possible testing techniques, assuming rather rigorous test conditions are applied. These techniques are discussed individually in the following subsections. Test sensitivity is indicated in terms of the minimum leakage rate detectable from a single leak, since percentage leakage rates would have no meaning without relating the specific containment vessel size and pressure-temperature conditions. The units indicated, cubic feet per day, were chosen to simplify subsequent calculations of percent leakage rate per day, if desired. Since variations in these sensitivities by one to two orders of magnitude can easily be caused by varying the test conditions, Table 10.2 should be used primarily as an indication of the relative sensitivities of the various techniques listed rather than as an indication

Table 10.2. Order of Magnitude Sensitivity of Various Local Leak Testing Techniques

| Technique | Typical Flow Detectable Under Specified Conditions ($ft^3$ per day) | Basis of Indicated Value[a] |
|---|---|---|
| Bubble observation tests | | |
| Soap bubble test | 10 | Observation of 2-in.-diam bubbles forming in 4 sec |
| Water submersion test | 0.01 | Observation of 1/16-in.-diam bubbles at one per second |
| Vacuum test | 0.1 | 10-$ft^3$ chamber; 1/2-hr test; constant temperature; pressure readable to 0.1 mm Hg |
| Sonic tests | 15 | |
| Adulterant gas tests | | |
| Air-ammonia test with HCl solution or phenolpthalein indicator | 1 | Ammonia concentration of $10^{-3}$ parts by volume |
| Halogen sniffer test | $10^{-3}$ | Instrument sensitivity of $1 \times 10^{-6}$ cc/sec; halogen concentration on pressurized side of $10^{-2}$ parts by volume; all leakage ducted to instrument with no external dilution |
| Helium mass spectrometer test | $10^{-6}$ | Mass spectrometer sensitivity of $5 \times 10^{-8}$ cc/sec; pure helium on pressurized side; all leakage ducted to the spectrometer with no external dilution |
| Radioactive gas test | 10 | 10-$ft^3$ chamber, 330 μc $^{85}Kr$ (1 mr/hr at 1 ft, unshielded) |
| Olfactory test | 1 | Average human sensitivity to mercaptan = $4 \times 10^{-8}$ parts by volume; local test mercaptan concentration of $10^{-3}$ parts by volume |

[a]See discussion of each of these tests for additional details on the assumed test conditions.

of the absolute sensitivities obtainable in every case. Obviously, the sensitivity of adulterant gas tests can be increased by increasing the concentration of the gas.

Selection of a suitable testing technique for any particular application will depend upon many factors other than the minimum sensitivity achievable, such as the equipment required, the time required, any hazards that may be involved, the worker skills required, and the test sensitivity required. It should be noted that detection of a leakage rate of 1.0 $ft^3$ per day from an individual leak, a sensitivity easily attainable by nearly all local testing techniques, would give a fractional leakage rate of 0.0001% per day per leak on a 1,000,000-$ft^3$ containment system. Allowing over 1000 leaks at rates lower than this sensitivity would not exceed the often specified 0.1% per day maximum initial-acceptance leakage rate. However, it must be realized that the original test requirements include a safety factor for errors and long-term deterioration.

Although hypothetical, this illustration provides an insight into the feasibility of using local leak testing as a means of continuous leak monitoring. Calibration of the leak technique, use of standard leaks as checks on calibration, grouping of penetrations into a single collection or testing system, and other techniques could be used to implement the method so that the leakage rate monitored could be expected to reflect the integrated leakage rate with reasonable confidence.

The following discussion of each of the local leak testing techniques describes the basic principle, discusses the accuracy and complexity, and indicates the equipment and worker skills required for each technique. Most of these techniques are also discussed in Appendix A of the proposed ANS leakage-rate testing standard.[1]

## 10.3.2 Bubble Observation Tests

### 10.3.2.1 Soap Bubble Test

The soap bubble test is the most common local leak detection method. In this test, as outlined in the proposed ANS Standard, the structure is pressurized to at least 5 psig (or half the design pressure, if less than 10 psig), and a soap solution is applied to the outer surface so that any leaks are detected by the presence of soap bubbles on the surface. This method is commonly used to make a complete survey of all welded joints, gaskets, threaded connections, and mechanical closures, such as air locks, valves, and pressure-release devices, while the containment vessel is being pneumatically pressurized for its initial pressure test. It is not useful in tests of components, such as valve seats, having complex geometries not readily exposed to soaping and visual observation operations.

In a typical soap bubble test, pressurization of the vessel is stopped after a pressure of approximately 5 psig is reached, and the soap solution is applied to all welds and joints on the entire external surface. Particular attention is given to locations where leakage might be expected to occur. If any leaks are detected at this pressure, they are repaired, with the vessel depressurized if necessary. Gasketed-type closures can be sealed off for later repair without depressurization provided the repaired part can be locally tested. When no further leaks are found, pressurization of the vessel is continued in increments until the test pressure is reached. After a 1-hr holding period at the test pressure, the pressure is reduced to design pressure, at which time a second soap bubble test is conducted. This test not only confirms results of the initial soap bubble test, but indicates whether additional leaks were opened up during the high-pressure strength test. If any significant leaks are detected during the test, they must be repaired prior to proceeding with the integrated leakage-rate test. If the soap bubble test is conducted carefully and all welds, joints, access openings, penetrations, etc. of the containment vessel are covered, the vessel will likely pass the integrated leakage-rate test because the sensitivity of the soap bubble test for small leaks can be greater than that of the integrated leakage-rate test. There is therefore a strong economic incentive to use the soap bubble test and thereby avoid an expensive and prolonged series of integrated tests. A possible difficulty with the soap bubble test is that large leaks might not be detected, since the air could break through the soap solution without forming bubbles. However, a leak that is large enough to break through the solution without causing bubbles should be detectable by a competent inspector either audibly or by sense of touch.

In another application of local leak testing, the soap bubble test is used in conjunction with a vacuum box. In this case, a vacuum box containing a window is placed over the area to be tested and is evacuated to produce at least a 5-psi pressure differential. The soap solution is applied to the test area before it is covered with the vacuum box, and any leaks will be indicated by bubbles observed through the window in the box. This method is particularly useful when the entire containment vessel cannot be pressurized, either prior to completion or after equipment which is pressure sensitive has been installed within the vessel. It is also used for retesting localized areas to avoid having to repressurize the vessel.

The equipment needed for the soap bubble test is simple, consisting of that needed to provide a pressure differential across the test surface, a soap solution, paint-brush applicators, access equipment, such as ladders, bosun's chairs, etc., a chronometer, and observant workers. The use of a controlled leak to demonstrate the principle and sensitivity of this test to unskilled workers is suggested.

A suitable soap solution for air leakage indication, as suggested in the proposed ANS standard for leakage-rate testing,[1] is one consisting of equal parts of corn syrup, liquid detergent, and glycerin. The solution should be prepared not more than 24 hr preceding the test, and its bubble formation properties should be checked with a sample leak every half hour during the test. Observation of a 2-in.-diam bubble forming in 4 sec will assure the 10-$ft^3$ per day sensitivity indicated. Other soap solutions could be used as well, the principal requirement being that sufficiently stable bubbles be formed that they can be detected visually for some period of time after they initially emerge.

The soap bubble test has been used to detect leaks in many types of pressure-containing systems and components and is an accepted and sensitive method of leak detection when properly applied. Although it may seem rather tedious and time-consuming to cover the weld seams, penetrations, etc. of the entire vessel, it is probably more efficient than possible alternative procedures. In the report of the leak tests performed on the Belgian reactors BR 2 and BR 3,[8] it was indicated that ten three-man teams performed a complete soap bubble test in less than 6 hr. A conclusion of the report was that "the qualitative soapy water test is very sensitive and rapid."

#### 10.3.2.2 Water Submersion Test

The water submersion test is a common and simple alternative to the soap bubble test that can be used for small components. This test is performed by establishing a differential pressure across the surface to be tested and covering the low-pressure side of the surface with water. A sufficient volume of water should be provided to permit full submergence of the surface and convenient observation of bubble formation. Adequate lighting must be available and sufficient time and varying positions taken to assure that bubbles are not collecting unseen in pockets of the testing device. Trapping the bubbles by covering the water with a clear plastic sheet will retain the bubbles and enable the inspector to see them more readily. Observation of 1/16-in.-diam bubbles appearing at a rate of one per second will indicate a leak of 0.01 $ft^3$ per day.

### 10.3.3 Vacuum Test

In the vacuum test method, a vacuum box is placed over the area to be tested and evacuated to at least a 5-psi pressure differential. A manometer attached to the vacuum box will indicate leaks in the area tested by an increase in the vacuum box pressure. A quantitative indication of leakage rate can be obtained by measuring the rate of change in the level of the manometer. A 0.1-mm-Hg decay of vacuum pressure in a period of 1/2 hr for a 10-$ft^3$ box at constant temperature will indicate a leak of 0.1 $ft^3$ per day. Considerable worker experience is likely to be necessary, especially in assuring leaktightness of the mating joints between the vacuum box and the test specimen, since leaks at the joints would totally invalidate any experimental results. The use of a controlled leak to verify the experimental techniques being used is suggested.

This method is particularly useful when it is not possible to completely pressurize the containment vessel. It may be most useful as a permanent installation at areas, such as penetrations, where leakage might be expected to develop. Concern about leakage of the box itself could be reduced in the permanent installation. This technique is discussed in Section 10.4.10 as a possible method of determining the total containment vessel leakage rate. The equations applicable are the same as those used in the absolute method for integrated leakage-rate testing outlined in Section 10.4.8.

### 10.3.4 Sonic Tests

During the test of the Plum Brook reactor containment vessel, it was demonstrated that fairly minute leaks could be heard, and immediately repaired, by operators who entered the vessel through the air locks while the vessel was at a low testing pressure.[9]

Several types of devices utilizing sonic or ultrasonic waves have been developed and used for detecting leaks in various pressure and vacuum systems, although they have not been used to any degree in containment vessels. One such device uses a probe that detects the ultrasonic waves generated by air rushing through a leak and produces a proportional signal in the audible frequency range that is transmitted to a set of earphones.[10] Because the noises are detected in the ultrasonic range, this method is quite insensitive to background audible noise. Since the intensity of the sound is inversely proportional to the square of the distance from the source, the leak can be located accurately by moving in the direction of increasing sound intensity. This device is small and can be carried easily. Leakage rates as low as 0.01 cfm can be detected up to distances of 10 ft, so this method could be quite valuable as an initial check for containment vessel leaks. Quantitative interpretation of results is not likely to be particularly accurate.

Another sonic device that has been used to locate leaks in pipelines uses an ultrasonic generator to produce a signal within the pipeline. At a leakage path, the ultrasonic signal is transmitted through the pipe wall and can be detected by transducers. Location of the leak path is determined by traversing the transducers along the outside of the pipes.

Either of these methods may be suitable for containment vessel leak detection, particularly if the vessel surface is no longer accessible and the soap bubble method cannot be used. The sensitivity of these sonic methods is comparable with that of the soap bubble test, and in addition they can be used to locate leaks that are too large to be detected by the soap bubble method. However, to be detected, the leak mechanism must involve gas velocities sufficient to generate ultrasonic waves, and the system geometry must be such that the ultrasonic waves so generated are not damped out by interferences from interposed materials, such as shielding, insulation, or structural supports.

Because of these qualifications, such tests, although they provide useful leak detection information, should not be relied upon as absolute detection mechanisms.

## 10.3.5 Adulterant Gas Tests

### 10.3.5.1 Air-Ammonia Test

The air-ammonia test is performed by adding anhydrous ammonia to the air-pressurized system. The outer (low-pressure) surface of the vessel is then probed with a swab wetted with 0.1 N hydrochloric acid. The presence of a leak is indicated by a white chemical fog that appears when the ammonia reacts with the hydrochloric acid.

Another indicator for use with ammonia is a phenolphthalein solution in equal parts of water and ethyl alcohol. A small cloth lightly dampened with this solution and placed over the test area will indicate a leak by a pink discoloration.

Both of these tests require rather high ammonia concentrations, and it is suggested that their use be limited to rather small test specimens. Both are only qualitative in nature, providing no significant quantitative information.

### 10.3.5.2 Halogen Sniffer Test

The halogen sniffer test is somewhat similar to the air-ammonia test in that an easily detectable halogen gas, such as freon, is added to the pressurized side of the surface to be tested and leaks are detected by scanning the low-pressure side of the surface. Leakage is detected by traversing the test area with a detector that senses the effect of the halogen compound on ion emission from a heated metal surface. Another method of detecting the halogen gas is by use of the flame coloration technique. Instruments able to detect 1 ppm are commonly available. Assuming a halogen concentration of $10^{-2}$ parts by volume and complete ducting of all leakage to the instrument without substantial external dilution gives a sensitivity equivalent to less than 0.001 $ft^3$ per day.

One difficulty with the use of halogens is that halogen detectors are also sensitive to cigarette smoke or dry-cleaning-fluid vapors. Also, the possibility of chloride stress corrosion of stress corrosion-sensitive materials inside the containment vessel should be considered. If halogens are used with these materials, their surfaces should be protected or thoroughly cleaned following the test. If freon is used in continuous monitoring systems, precautions should be taken to avoid radiolytic decomposition into hydrochloric and hydrofluoric acid.

The requirement of having rather high internal halogen concentrations suggests limiting this technique to rather small tests to avoid the hazards of handling larger quantities of toxic materials by workers who may be relatively unfamiliar with toxicity problems. Additionally, considerable skill and practice with a controlled leak should be obtained before attempting to ascribe quantitative significance to experimental results.

### 10.3.5.3 Helium Mass Spectrometer Test

Another leak detection technique, similar to the halogen method, uses helium as the detectable gas introduced into the pressurized air. Relatively high helium concentrations, say $10^{-2}$ parts by volume, can be used without fear of toxicity problems, and thus the test may be used for large as well as small applications. The price of the helium, normally less than \$50 per 1000 $ft^3$ (plus delivery), should be relatively insignificant. Mass spectrometers that can detect helium with a sensitivity of $5 \times 10^{-8}$ cc/sec are available. This instrumentation, when used with pure helium on the pressurized side, and the ducting of all leakage to the instrument without additional dilution gives a sensitivity equivalent to $10^{-6}$ $ft^3$ per day.

Although this degree of sensitivity is usually not required for testing large containment vessels, a helium test is often used to determine the leaktightness of individual components that cannot be conveniently soap bubble tested[11] and of the reference system before conducting a leakage-rate test using the reference vessel method, as described in Section 10.4.9. A relatively high degree of worker skill is required, and considerable experience with controlled leaks should be available before quantitative interpretation of experiment results is attempted.

A helium leak test of all welded joints was performed on the SM-1 containment vessel.[12]

#### 10.3.5.4 Additional Adulterant Gas Methods

Additional testing methods include the detection of adulterants such as $^{85}Kr$, which can be detected by its radioactivity, or of gaseous additives that have particularly strong odor characteristics, such as $SO_2$ or mercaptan-based compounds. Care must be exercised in the use of these methods. The safe use of radioisotopes requires a knowledge of radiation protection standards and techniques, and the safe use of odorous gaseous additives requires a knowledge of their toxic characteristics. Because these problems tend to increase with test size, both of these techniques are likely to be more useful in bench-scale testing of accessory components than they will be for integrated system leak testing. With experience, the radioisotope adulterant gas method should yield good quantitative information. However, the sensitivity of humans to odors is sufficiently variable, especially after repeated exposures, that quantitative interpretation of odorant tests should not be attempted, except possibly on a "go—no go" basis.

## 10.4 LEAKAGE-RATE TESTING

### 10.4.1 General

Leakage-rate testing is performed to determine the integrated leakage rate of air out of a containment vessel at design pressure in order to demonstrate that the measured rate does not exceed the initial acceptance leakage rate specified for the vessel. In specifying the acceptance value and the maximum allowable leakage rate, the consequences of an assumed maximum credible accident are analyzed, taking into account the amount and type of radioactive materials that could be released to the containment, deterioration of the containment, the possible mechanisms for their dispersal, and the potential harm to the surrounding population. Because of the many uncertainties associated with the release and transport of radioactive materials during the assumed accident, several assumptions must be made in the safety analysis to justify leak testing as now performed. For example, it is generally assumed that the fission products released from the reactor will leak out of the contain vessel at the same rate as in air, whereas during an accident, considerable condensation, deposition, and filtration of the fission products probably would occur in the vessel and in the minute leakage paths. (The generation and transport of radioactivity during an accident and the assumptions normally made in safety analyses are discussed in detail in Chapter 4.) Other conditions, such as temperature and humidity, will also be different during a leakage-rate test than in the accident situation, since simulating them during the test may not be practical or justifiable in view of the other assumptions involved. Pressure is the most significant variable in determining the leakage rate, but in many cases it is necessary to conduct a reduced pressure test and extrapolate the measured leakage rate to an equivalent rate at the design pressure, neglecting or estimating any changes in leakage path size with changes in pressure. The leakage rate determined in a containment vessel leakage-rate test is, therefore, only an approximation of the rate at which fission products would leak out of the vessel in the event of the assumed accident. The intent is that the leakage-rate test be conservative, along with all other factors that demonstrate containment capability. Consideration must therefore be given to the conditions existing at the time of the postulated mca in order to have assurance that conservatism does in fact exist.

It has been the practice with many containment vessels to conduct both an initial and a final integrated leakage-rate test before the plant is operated. The initial test is performed as soon as the bare containment shell has been completed and with the penetration openings blanked off. It is conveniently performed at full design pressure immediately after the pressure test and is intended to demonstrate the leaktightness of the shell itself, primarily to show that the vessel fabricator has fulfilled his contractual obligations. The final test is performed after all penetrations and equipment have been installed; it demonstrates the leaktightness of the entire containment system. In some instances this final test has been conducted at reduced pressure because of possible damage to some of the installed equipment. In such cases, it may be that neither the initial nor the final test has been considered a completely adequate test by itself and both may be cited as evidence of leaktightness. This practice is controversial and may not be one that can be justified for most circumstances. In view of the present uncertainty of the precise effect of accident conditions on fission-product transport and leakage and also in view of the gravity of excessive release from the large reactors now being proposed, a present conservative viewpoint is to employ final and retest procedures that determine the leakage rate as a function of pressure up to design pressure or to employ continuous monitoring systems or both. The techniques for performing the initial and final leakage-rate tests are essentially the same, and no distinction between the two tests is made in the remainder of this section. Notwithstanding the similarity of technique, the employment of testing equipment may be greatly different as a result of an internal void for the initial test and the inclusion of process equipment, concrete, compartments, etc. for the final test and retests.

This section discusses leakage-rate testing primarily as related to conventional steel-shell containment vessels. Because of greater experience with this type of containment structure, there is considerable agreement, as reflected in the proposed standards, on the procedures to be used for leakage-rate testing. Many or all of these procedures may be applicable for other types of containment structures as well, but more consideration of the special conditions involved may be required to ensure that the results are valid. Section 10.4.13 presents a brief discussion of some of the considerations that may apply to the leakage-rate testing of various types of containment structures.

### 10.4.2 Leakage-Rate Criteria

Since reactor containment is provided to prevent the release of appreciable amounts of radioactivity in the event of a nuclear accident, the allowable leakage rate must be based on what is acceptable for nuclear safety considerations. Reactor type and size, site location and meteorology, as well as the possible mechanisms for radioactivity generation and transport within the containment vessel, are all considered in specifying the maximum allowable leakage rate for a given containment system. Hence, leakage-rate criteria cannot be standardized but must be established individually for each plant. Because of the general desire to reduce nuclear hazards to the practical minimum, the lowest leakage rate that is readily obtainable and measurable is often specified, even though it may be much lower than is required by hazards considerations.

For the conventional single-barrier steel-shell type of containment used for most of the power reactors built thus far, leakage rates well below initial acceptance values for nuclear safety considerations have been demonstrated. A maximum leakage rate at design pressure of 0.1% by weight of contained air per day has been specified for several vessels of this type, and, when the vessel has been designed and built with careful attention to penetrations, welds, and other details, the specified rate has been achieved without great difficulty or expense. Consequently, a leakage rate of 0.1% per day is almost considered a standard initial leakage rate for this type of containment. Attaining this low initial leakage rate results in part from experience with bare shells and may not be practical for greatly compartmented or otherwise encumbered containment structures. Although leakage rates as low as 0.05% per day appear to have been demonstrated, leakage rates much less than this (say 0.01% per day or less) may be very difficult to demonstrate for single-barrier containment systems because of the limited sensitivity of normal leakage-rate testing methods, even though it might be possible to actually achieve such a leakage rate by careful design and construction.

For concrete containment structures with welded steel liners to provide leaktightness, leakage rates comparable with those of steel vessels should be achievable if the liner is designed and fabricated with the same care and attention to detail as are steel vessels. Unlined concrete structures cannot approach welded steel vessels in leaktightness and are seldom used where there is a requirement for a low leakage rate. Concrete is inherently porous and subject to cracking and is difficult to seal around penetrations. Even with wall thicknesses as great as 4 to 5 ft and with extra care taken to obtain high density and high strength, a concrete containment structure constructed by conventional means will leak several percent per hour.[6] When a nonmetallic coating is applied to the inner surfaces, leakage may be reduced by a factor of 2 to 3, but if the coating is subject to cracking, it may have little effect.[6] If special joints, penetrations, and coatings are used, it may be possible to construct large concrete buildings for design pressures up to 5 psig with leakage rates as low as 1% of their contained volume per day.[13] Since the aging of a concrete structure can considerably reduce its leaktightness, frequent periodic retests may be a licensing requirement.

For composite concrete and steel containment structures, consisting of a steel dome attached to a concrete base, leakage is likely to occur through the concrete, through penetrations in the concrete, and through the joint between the concrete and steel. The HWCTR composite containment vessel, containing a thermosetting plastic resin on all interior concrete surfaces and a special joint between the concrete and the steel shell, had a leakage rate of about 0.5% per day at 24 psig.[14] The concrete portion of the vessel was prestressed to reduce the number and size of cracks that might develop.

Multiple-barrier containment designs may be required if hazards considerations dictate lower maximum allowable leakage rates or more positive control of leakage than can be achieved even by steel single-shell containment vessels. In a two-barrier containment system, leakage from the primary vessel may be collected within the secondary containment boundary and discharged to a stack or returned to the primary containment after being passed through a cleanup system. This not only removes a large part of the radioactivity from the leakage but also assures that it will be released at the top of a stack rather than at ground level so that atmospheric dilution will have a greater effect. If the secondary containment space does not contain a cleanup system or if the system is inoperative, containment effectiveness is probably still considerably improved because of the additional holdup this space provides for any leakage from the primary vessel. Anderson[15] and Rushton and Edwards[16] have developed equations and curves to evaluate the effectiveness of systems of this type. Discussion of some of this work is covered in Section 4.4.4.5.

The primary containment vessel in a double containment system is normally a welded steel shell with the maximum leaktightness attainable with existing containment construction and inspection techniques. The secondary barrier may have any degree of leaktightness, depending upon the other functions of the structure that provides this barrier and the degree of overall containment effectiveness required. The NS Savannah utilizes a ship's compartment as a secondary containment barrier. The concrete containment shield of the Indian Point plant can be used to provide secondary containment. In the containment system that was proposed for the Ravenswood plant in New York, welded steel shells were used for the primary and secondary boundaries and any leakage through either shell was pumped back into the primary containment vessel to provide positive control of leakage to the negative pressure zone between barriers in the event of an accident.[7] In systems of this type, separate maximum allowable leakage rates must be specified for each barrier such that the overall leakage rate meets the leakage-rate criteria established for the plant. Double containment systems are suitable for leakage-rate testing using adulterant gas methods (see Sec. 10.4.10.2).

Pressure-suppression containment systems are essentially like other types of containment in that they have a static envelope completely surrounding the nuclear system. A difference is that the dry-well envelope usually will have a higher design pressure and smaller volume than other types of containment. The suppression chamber will have a lower design pressure or smaller volume or both. The water in the suppression chamber may be considered to be a second containment barrier, since it may trap some of the fission products, except the noncondensable gases. In some pressure-suppression containment designs, the suppression chamber is contained in a concrete vault, which serves as another containment barrier. A lower allowable leakage rate may be specified for the dry well than for the suppression chamber, since the dry well would contain considerably more airborne radioactivity. If it is assumed that the fission-product release is coincident with the initial pressure peak in the dry well, an even more stringent allowable leakage rate for the dry well is required. However, testing the two vessels separately after the plant has been completed is difficult because they are interconnected. At Humboldt Bay, the dry well and the suppression chamber were leakage-rate tested separately during construction of the plant. A continuous leakage-rate monitoring system is proposed for checking containment integrity during operation (see Sec. 10.4.10.1).

An underground concrete containment system has been proposed[17] in which the ground-water pressure exceeds the maximum pressure produced in an accident. Leakage-rate testing of such a system would be more sensitive than for many conventional types of containment as a result of a lack of diurnal temperature swings. In addition, the always present external pressure would provide a high degree of assurance of leaktightness without dependence on a mechanical system. Reliance upon the external pressure would require continuous monitoring to be assured of positive control.

For containment systems other than the conventional types, there are no standards or established practices for leakage-rate testing. Testing methods for these systems must be established on a case basis to meet, to the extent possible, the intent of the established standards. It may take considerable ingenuity to demonstrate very low leakage rates, even for structures that are inherently quite leaktight. In specifying a maximum allowable leakage rate it is important to consider how the leakage-rate test will be conducted and what the significance of the results will be.

The effectiveness of nearly all types of containment depends on the proper operation of certain mechanical devices and systems. For example, the ventilation system in many containment systems discharges up the stack during normal operation and depends on the operation of isolation valves to close the system in the event of an accident. In plants in which the main steam line penetrates the containment boundary, steam line isolation valves and turbine stop valves must close reliably to isolate the plant. In some multiple-barrier designs, in addition to the need for closing the normal ventilation system, blowers must continue to maintain a reduced pressure in the annular space if the full benefit of this system is to be realized. Pressure-relief containment systems, which vent the initial pressure surge during an accident and then close before fission products are released from the core, are dependent on the proper timing and operation of a mechanical device, as well as on an adequate knowledge of the rate of release of fission products. It is evident that the careful specification, design, and testing of the mechanical systems that form an integral part of the containment system are as important as specifying the maximum allowable leakage rate and demonstrating that it can be achieved. Without proper operation of these systems, containment vessel leakage-rate criteria are meaningless. These isolation systems are discussed in Chapter 9.

### 10.4.3 Terminology

The terminology applied to the field of leak detection and leakage-rate measurement is not always consistent. In this chapter, the terminology generally follows that used in the proposed ANS leakage-rate testing standard.[1] The following definitions are used:

1. Leak. A leak is an opening that allows the passage of a fluid.
2. Leakage. Leakage is the fluid that escapes from a leak. For containment vessels, the reference fluid is air.
3. Leakage Rate. The leakage rate is the leakage experienced during a specified period of time. In this chapter, leakage rate is the percent by weight of the total amount of air contained in the vessel at initial test conditions that leaks out in 24 hr.
4. Maximum Allowable Leakage Rate. The maximum allowable leakage rate is the maximum leakage rate specified in the Operating License and Technical Specifications for a particular containment structure.

From the context of these definitions, the test leakage rate, since based on air, must have an established basis for extrapolation to the accident fluid composition and state in order to confirm compliance. For example, if the leakage rate is determined from pneumatic pressurization of the containment vessel during a test, the leakage specification reported in the following manner will clearly define its meaning:

1. Design Leakage Rate

   _____ wt % loss of containment atmosphere in 24 hr for the postulated accident conditions

2. Measured Leakage Rate

   _____ wt % loss (±_____) of air in 24 hr, at an average temperature of _____ °F and a pressure of _____ psia as calculated from leakage test data

3. Extrapolated Leakage Rate (for tests conducted at other than containment design pressure)

   _____ wt % loss (±_____) of air in 24 hr at a temperature of _____ °F and a pressure of _____ psia

4. Corrected Leakage Rate

   _____ wt % loss (±_____) of steam-air mixture in 24 hr at a temperature of _____ °F and a pressure of _____ psia with a ratio of steam to air of _____, corresponding to the calculated conditions of the postulated loss of coolant accident

Some ambiguity and confusion has resulted when the leakage rate has been specified only as "percent per day." As Brittan[18] has pointed out, this could imply percent of vessel volume, percent of total contained air, percent of air added during pressurization ("stored air"), or percent of design pressure. If leakage rate is specified on a volume basis, the temperature and pressure must be clearly specified also. It is not clear whether using total contained air or stored air as a basis is more representative of the accident condition. However, the difference is usually not significant, in view of the other approximations made in safety analyses, provided the basis is understood and clearly stated when specifying the allowable percentage leakage rate and provided the measured rate is determined on the same basis.

For the EBWR the leakage rate was specified as 500 $ft^3$ at standard temperature and pressure per 24 hr for the initial test of the empty containment shell (approximately 0.1% of the "stored" air) and 1000 $ft^3$ (STP) per 24 hr after installation of internal concrete and equipment (approximately 0.25% of the "stored" air in the free volume).[18] For most other plants the basis has not been clearly stated and no distinction has been made between allowable rates for the empty shell and the completed plant. It has usually been the case, however, that the leakage rate is determined as a percent by weight of total contained air, and the same allowable rate has been set for both initial and final leakage-rate tests. The amount (weight or volume) of air leaking out per hour, which will be different for the bare shell and for the completed plant at a given percentage rate, has not usually been determined, since it is the fraction of contained radioactivity leaking out that is most significant in the safety analysis. The previous ambiguities and difficulties in comparing rates for different plants should be largely eliminated by the adoption of the proposed ANS leakage-rate testing standard.[1]

## 10.4.4 Standards

Although leakage-rate testing of the containment vessels built to date has not been performed in accordance with any accepted standard, the methods used have usually been quite similar and are reflected in the two proposed standards that deal with this subject.

### 10.4.4.1 ANS Proposed Standard

The proposed Standard for Leakage-Rate Testing of Containment Structures for Nuclear Reactors,[1] which was prepared by the ANS-7 Subcommittee of the ANS Standards Committee, reflects the practice which has generally been used in the past and which might be expected to be followed in the future. The provisions of the proposed standard apply "to containment structures for nuclear power, test, research, and training reactors, wherever a gastight containment structure is specified as a condition for operation." Because of its general applicability and its reflection of current industrial practice, this standard will be referred to throughout this section. The following provisions are quoted from the standard:

> "4.1 Sequence of Tests. Leakage-rate testing should be conducted after the inspection and testing of welded joints, penetrations, and mechanical closures; completion of repair measures for the minimizing of leakage; and completion of containment structure pressure tests for strength. Where the

containment structure is to be subsequently covered with concrete or will otherwise be inaccessible to direct examination, particular care should be given to inspection of these areas prior to such coverage. Integral or local leak detection should preferably precede leakage-rate tests.

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"5.1 Applicable Test Methods. Leakage-rate test procedures applicable to this standard may be either the absolute method or the reference vessel method. The choice of either method shall be a matter of agreement between parties who are charged with responsible acceptance of the vessel and those in charge of the leakage-rate test procedures.

"5.2 Description of Methods. The absolute method of leakage rate testing shall constitute the determination and calculation of air losses by containment structure leakage over a stated period of time by the means of direct pressure and temperature observations during the period of test with temperature detectors properly located to provide an average air temperature. The reference vessel method shall constitute the determination and calculation of air losses by observations of the pressure differentials between the containment structure and a gastight reference system, with the reference vessels located so as to represent, with reasonable accuracy, the average temperature of the aggregate containment air.

"5.3 Leakage-Rate Pressures. Leakage-rate determinations shall be conducted at the pressure at which the leakage rate was specified and after all pressure testing required in the specification.

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"7.1 The Absolute Method. The absolute method of leakage-rate determination depends on the measurement of the temperature and pressure of a constant volume of containment structure air with suitable correction for changes in temperature and humidity control under a nearly constant pressure difference with respect to the atmosphere outside the structure. It is assumed that the temperature variations during the test will be insufficient to effect significant changes in the internal volume of the structure or the partial pressure of water vapor in the contained air.

"7.2 The Reference Vessel Method. The reference vessel method of leakage rate determination depends on the changes in pressure of a constant volume of contained air compared with that of a hermetically closed reference vessel which may be at the same pressure as the contained air at the start of the test or may have a small differential. The reference vessels shall be so placed and of such a geometry that they will assume the temperatures of the contained air within a reasonable time lag. The reference vessels shall be subject to leakage rate determination in accordance with the absolute method prior to their use for containment structure testing according to the applicable procedures of this standard or may be checked by the halogen sniffer test or by retention of vacuum.

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"7.7 Period of Test. The leakage rate test period shall extend to not less than 24 hr of retained internal pressure. Completion of the test should be scheduled to coincide with atmospheric temperatures and pressures close to those at the start of the test, as far as is possible. Check tests or repetition of tests shall be a matter of agreement between those responsible for the acceptance of the containment structure and those in charge of the leakage-rate testing."

10.4.4.2 ASA Proposed Standard

The leakage-rate testing provisions of the proposed ASA N6.2 Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors (Appendix E), although in less detail and somewhat more limited in application, are generally consistent with the provisions of the ANS Standard. The ASA Standard is listed as one of the conjunctive standards in the ANS Standard. The following provisions are quoted from Section 16, Leakage Testing, of the ASA Standard:

"1. Both local leak tests and integrated leakage-rate tests shall be performed. The local tests are to locate leaks of detectable size. The integrated leakage-rate test is to make sure that the overall leakage-rate does not exceed the allowable value.

"2. The initial integrated leakage-rate test consists of pneumatically pressurizing the shell to between 85% and 100% of its design pressure and observing changes in pressure with time. Pressure shall be measured with sufficient accuracy over a long enough period of time to insure that the leakage is less than the allowable leakage rate. Allowances must be made for the effects of temperature and humidity variations on the pressure readings.

"3. Either of two methods may be used for leakage rate testing:

Pressure, temperature, and humidity measurements at a number of points inside the containment shell shall be taken over a long enough period of time to insure reliability.

Pressure changes in the containment shell may be measured with respect to a reference system of one or several containers distributed within the shell. At the beginning and end of the test period, the average temperature of the air in the reference system and in the containment shell shall be approximately equal.

"4. Penetrations and seals shall be subjected to local leak tests and shall normally be exposed to the integrated leakage-rate test. When there are two seals in series on a penetration, then one seal shall be open during the leak-rate test and the other must be closed. The integrated leak-rate test need not be repeated with the open-and-closed situation reversed if both the seals in the series have successfully withstood the pressure test and revealed no imperfections in the local leak tests. Minor penetrations added as a result of new work shall be subjected to local leak tests but need not be subjected to integrated leakage-rate tests."

## 10.4.5 Nature of Leakage

### 10.4.5.1 Theoretical

The nature of the leakage, including the equations that define its rate as a function of its driving forces, is dependent upon a number of factors. The major factors of importance are whether or not the critical pressure is exceeded, the diameter, length, and roughness or tortuosity of the leakage path, and the mean free path and viscosity of the escaping gas. Various authors have classified different forms of leakage according to the specific physical phenomena involved, some of which may exist only over a certain range of conditions. The following list, which is not meant to be all inclusive, represents the major categories normally considered:

1. Molecular Diffusion. Molecular diffusion is flow caused only by a varying concentration gradient. This type of flow is independent of pressure drop other than as it affects the resulting concentration gradient.
2. Molecular Flow. In molecular flow, the flow path cross section is small relative to the mean free path of the gas molecule so that the gas molecules collide with the walls more often than with each other.
3. Viscous, Laminar. In viscous, laminar flow, the flow path cross section is large relative to the mean free path of the gas molecules so that the gas molecules collide with each other more often than with the walls, thereby supporting viscous stresses, and the flow velocity is sufficiently low to provide laminar flow conditions.
4. Turbulent Flow. Turbulent flow is the same as viscous, laminar flow, but the flow velocity is sufficiently high that turbulence is caused. Both subsonic and supersonic conditions are included.

The basic symbols and nomenclature used for the discussion of flow modes are listed below. For consistency, the units for each symbol are given in terms of mass (M), length (L), temperature (T), and time (t).

| Symbol | Definition | Units |
|---|---|---|
| a | Radius of a leak of circular section or one-half of the short dimension of a rectangular-section leak | L |
| A | Cross-sectional area of leak path | $L^2$ |
| c | Sonic velocity of gas | L/t |
| $C_d$ | Coefficient of discharge (nozzles and orifices), subscripts de and dt refer to coefficients at extrapolated and test conditions, respectively) | Dimensionless |

| Symbol | Definition | Units |
|---|---|---|
| $c_p$ | Specific heat of gas at constant pressure | L/T |
| $c_v$ | Specific heat of gas at constant volume | L/T |
| D | Diameter of circular cross-section leak | L |
| f | Friction factor | Dimensionless |
| g | Gravitational acceleration | $L/t^2$ |
| h | Width of rectangular passage (leak slit) transverse to flow | L |
| H | Periphery of flow channel, perimeter of noncircular leak | L |
| $\kappa$ | Ratio of specific heats of gas $c_p/c_v$ | Dimensionless |
| K | Constants in mass-rate or leakage-rate equations; subscripts 1 through 5 and m, $l$, tr, ts, and o, refer, respectively, to molecular, laminar, turbulent (rough), turbulent (smooth), and orifice-type flow | Dimensionless |
| $l$ | Length of leak path (between pressure zones $P_a$ and $P_o$) | L |
| L | Leakage of a containment system (containment atmosphere mass basis) | Dimensionless |
| $L_a$ | Leakage rate of containment system at conditions a, which may correspond to a test condition | 1/t |
| $L_b$ | Leakage rate of containment system at conditions b, which correspond to the postulated accident conditions | 1/t |
| $L_c$ | Leakage rate of containment system at extrapolated conditions (based on air in vessel at same temperature as under test conditions) | 1/t |
| $L_r$ | Leakage rate of a containment system | 1/t |
| $L_t$ | Leakage rate of containment system at the test conditions (based on air in vessel) | 1/t |
| M | Molecular weight of gas | |
| n | Number of leaks | |
| $N_m$ | Mach number | Dimensionless |
| $N_r$ | Reynolds number | Dimensionless |
| P | Containment pressure | $M/Lt^2$ |
| $P_a, P_b$ | Upstream absolute pressure of a leak path (corresponds to the containment vessel pressure) | $M/Lt^2$ |
| $\overline{P}_a, \overline{P}_b$ | Upstream absolute pressure of a leak path; same as above except for units | Atmospheres, absolute |
| $P_{av}$ | Average pressure between the upstream and downstream pressure of the leak path | $M/Lt^2$ |
| $P_e$ | Containment vessel absolute pressure at extrapolated conditions (based on air in vessel) | $M/Lt^2$ |
| $\overline{P}_e$ | Same as above, except expressed in units of atmospheres, absolute | |

| Symbol | Definition | Units |
|---|---|---|
| $P_o$ | Downstream absolute pressure of a leak path; corresponds to the barometric pressure outside of a containment vessel | $M/Lt^2$ |
| $P_t$ | Containment vessel absolute pressure at test conditions (based on air in vessel) | $M/Lt^2$ |
| $\overline{P}_t$ | Same as above, except expressed in units of atmospheres, absolute | |
| $Q$ | Volumetric flow rate | $L^3/t$ |
| $Q_a$ | Molecular flow rate (in terms of pressure times volume divided by time) | $\mu$, $L^3/t$ |
| $r$ | Radius of leak of circular section | $L$ |
| $R$ | Gas constant (subscripts a and b refer to applicable values under conditions a and b | $L/T$ |
| $R_H$ | Hydraulic radius, A/H | $L$ |
| $t$ | Time interval ($\Delta t$ usually 24 hr for leakage-rate test) | $t$ |
| $T$ | Temperature of gas (subscripts a and b refer to temperature at conditions a and b | $T$ |
| $V$ | Total volume of gas or total free volume of containment | $L^3$ |
| $V_c$ | Velocity of gas in compressible flow | $L/t$ |
| $v_m$ | Average molecular speed | $L/t$ |
| $V_s$ | Average gas velocity | $L/t$ |
| $w_e, w_t$ | Mass rate of flow in terms of weight of gas flowing through leaks per unit of time at extrapolated and test conditions | $ML/t^3$ |
| $w_a, w_b$ | Same as above, except for conditions a and b | $ML/t^3$ |
| $W_1$ | Weight of gas in free volume of containment remaining after the loss sustained through leaks for the selected interval of time | $ML/t^2$ |
| $W_o$ | Weight of gas in free volume of containment vessel at initial selected state | $ML/t^2$ |
| $\lambda$ | Mean free path of gas molecules | $L$ |
| $\mu$ | Viscosity of gas | $M/Lt$ |
| $\mu_v$ | Pressure unit used in vacuum work, $\mu$ Hg | $M/Lt^2$ |
| $\rho_{av}$ | Average fluid density between the upstream and downstream pressures of the leak passage | $M/L^3$ |
| $\rho$ | Fluid density in a leak passage | $M/L^3$ |

A brief review of the parameters that govern the flow characteristic of a gas passing through a small restriction will help to clarify the problem of leakage behavior. Experimental determinations of the flow rate of gases through leaks have been made by Nerken[19] for the size of leaks of interest in vacuum work and leak detection. Since some leaks in containment structures approach the size of leaks investigated by Nerken, the results of his experiments and comparison with theoretical equations can be applied to evaluate the magnitude and significance of leakage. These investigations emphasize, however, the fact that the effects of pressure on the flow rate of leakage (at atmospheric pressure) are not always simply apparent, because a given leak can exhibit several types of flow.

For the particular characteristic geometry of a leak in a vessel, it is desirable to recognize the type of flow that may occur, since, in turn, it defines the expected leakage rate in terms of the following controlling conditions:

1. Geometry of the leak and passage roughness,
2. Pressure difference across the leak,
3. Physical properties of the gas (i.e., viscosity, molecular weight, density),
4. Composition of the gaseous mixture (air-steam mixtures).

In formulating the leakage parameters for each type of flow investigated, cognizance is taken of the fact that leakage from a containment structure is preferably expressed as the fraction (or percentage) by weight of the contained atmosphere escaping per unit of time. Accordingly, all equations that define flow rate will be expressed in terms of the weight of gas escaping through the leaks.

1. Molecular Flow. The molecular flow regime is investigated primarily to evaluate its contribution to containment leakage, although this flow is predominantly significant under vacuum conditions, which have not been encountered in the reactor containment structures built to date. The concept of the mean free path of a gas is introduced at this point to define the flow regimes in which molecular flow may exist in containment vessel leaks. The mean free path is defined as the average distance that a gas molecule will travel between collisions with another molecule and varies inversely as a linear function of the pressure. At atmospheric pressures, the mean free path of air is exceptionally small. The average value of the mean free path of air molecules (as reported by Dushman[20]) at standard conditions of 1 atmosphere and 70°F is of the order of $2.6 \times 10^{-6}$ in. and decreases to approximately $2.6 \times 10^{-7}$ in. at 10 atm. Molecular flow occurs when the mean free path of the gas exceeds the largest dimension of the cross-sectional flow area of the leak passage. Under this condition the molecules will collide more frequently with the walls of the leak passage than with each other.

This mean free path of air may be compared with the size of the leaks that exist in a containment vessel. The smallest dimension of leak size that is significant with respect to flow rate is considered to be approximately $1 \times 10^{-4}$ in.,* since leaks below this size do not contribute materailly to the loss of the containment vessel atmosphere.

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*This might be compared with a leak size of $1 \times 10^{-1}$ in., which under constant pressure of 40 psig, would leak at a rate of 0.1% per day based on orifice-flow leakage from a containment structure with a free volume of $2.5 \times 10^{6}$ ft$^3$.

Obviously, molecular flow does not play a major role in contributing to containment leakage because of the exceedingly small leak dimensions required in this flow regime. However, molecular flow may occur in portions of irregular leak passages where considerable dimensional changes occur, such as in tight cracks in weld seams. If a substantial number of these small leaks existed in a welded containment structure, some slight contribution to the overall leakage might be expected as a result of molecular flow. In the field of vacuum technology, a convenient parameter frequently used to determine the type of flow is the Knudsen number, $a/\lambda$, where a is the radius of the leak with a circular cross section, and $\lambda$ is the mean free path of gas molecules flowing through the leak passage. This parameter, as discussed by Dong and Bromley,[21] affords a convenient means to arbitrarily divide the flow regimes into the various types of flows that characterize the gradual transition between molecular flow and laminar flow.

The flow regimes have arbitrarily been divided into a range expressed by the value of the Knudsen number:

| | |
|---|---|
| Viscous (or laminar) flow | $a/\lambda > 10^2$ |
| Slip flow | $10^2 > a/\lambda > 10$ |
| Transition flow | $10 > a/\lambda > 10^{-1}$ |
| Molecular flow | $a/\lambda > 10^{-1}$ |

By considering the conditions in a containment vessel during testing with air at low pressures such that the mean free path of air molecules is $2.5 \times 10^{-6}$ in., an indication may be derived from Table 10.3 of the flow regime that applies to the range of leak sizes normally expected in a containment structure.

It is apparent from the values of Table 10.3 that the prevailing type of flow, for the specific conditions assumed, is in the viscous flow regime and that transitional or molecular types of flow are not of any significant consequence for the size of leaks of interest in containment vessels. Within the range of pressures to which containment vessels are subjected, flows other than molecular may be expected when the free path of the air (at test conditions) is considerably less than the mean cross-sectional dimension of the leak size.

Table 10.3. Knudsen Number Correlation of Flow Regime and Leak Size

| Leak Size, 2a (in.) | Knudsen Number, $a/\lambda$ | Flow Regime |
|---|---|---|
| $1 \times 10^{-1}$ | $2 \times 10^4$ | Viscous |
| $1 \times 10^{-2}$ | $2 \times 10^3$ | Viscous |
| $1 \times 10^{-3}$ | $2 \times 10^2$ | Viscous |
| $1 \times 10^{-4}$ | $2 \times 10$ | Slip flow |

A reasonable approximation (as derived from charts developed by Santeler and Moller[22]) of the relationship between pressure, the mean dimension of the leak, and the flow regime for air at an ambient temperature of 70°F is illustrated by Fig. 10.1. The principal zone of interest for the range of pressures and leakages in containment vessels is defined by the rectangular area a-b-c-d, which clearly delineates the predominance of viscous flow. At this point, it is well to emphasize that other than viscous flow may be possible for the conditions outlined, since other leak flow parameters may be introduced with variations of leak geometries and containment test conditions.

To provide a more comprehensive exposition of other parameters that govern in the molecular flow regime and to enable comparison with other types of flows, the molecular flow equation for the simplest of leak geometry (circular section) is selected. The molecular flow equation as originally derived by Knudsen (discussed by Dong and Bromley[21] and by Dushman[20]), for a finite cylindrical leak, in the case of air flow from a higher pressure, $P_a$ (containment pressure), to atmosphere, $P_o$, is

$$Q_a = \frac{4}{3} v_m \frac{A^2}{H} \frac{P_a - P_o}{l} , \qquad (10.1)$$

which expresses the mass flow rate, $Q_a$, in the mass flow rate units (PV/t terms) normally applied in vacuum work. The equation may be converted to express the flow rate, w, in terms of weight, for the case of a circular leak of diameter, D, by substituting

$$H = \pi D ,$$

$$Q_a = wRT ,$$

$$v_m = \left(\frac{8RTg}{\pi M}\right)^{1/2}$$

in Eq. (10.1) so that

$$w = K_1 \frac{A^2}{lD} \frac{P_a - P_o}{(MRT)^{1/2}} , \qquad (10.2)$$

or, substituting $(\pi/4)D^2$ for A,

$$w = K_1' \frac{D^3}{l} \frac{P_a - P_o}{(MRT)^{1/2}} , \qquad (10.3)$$

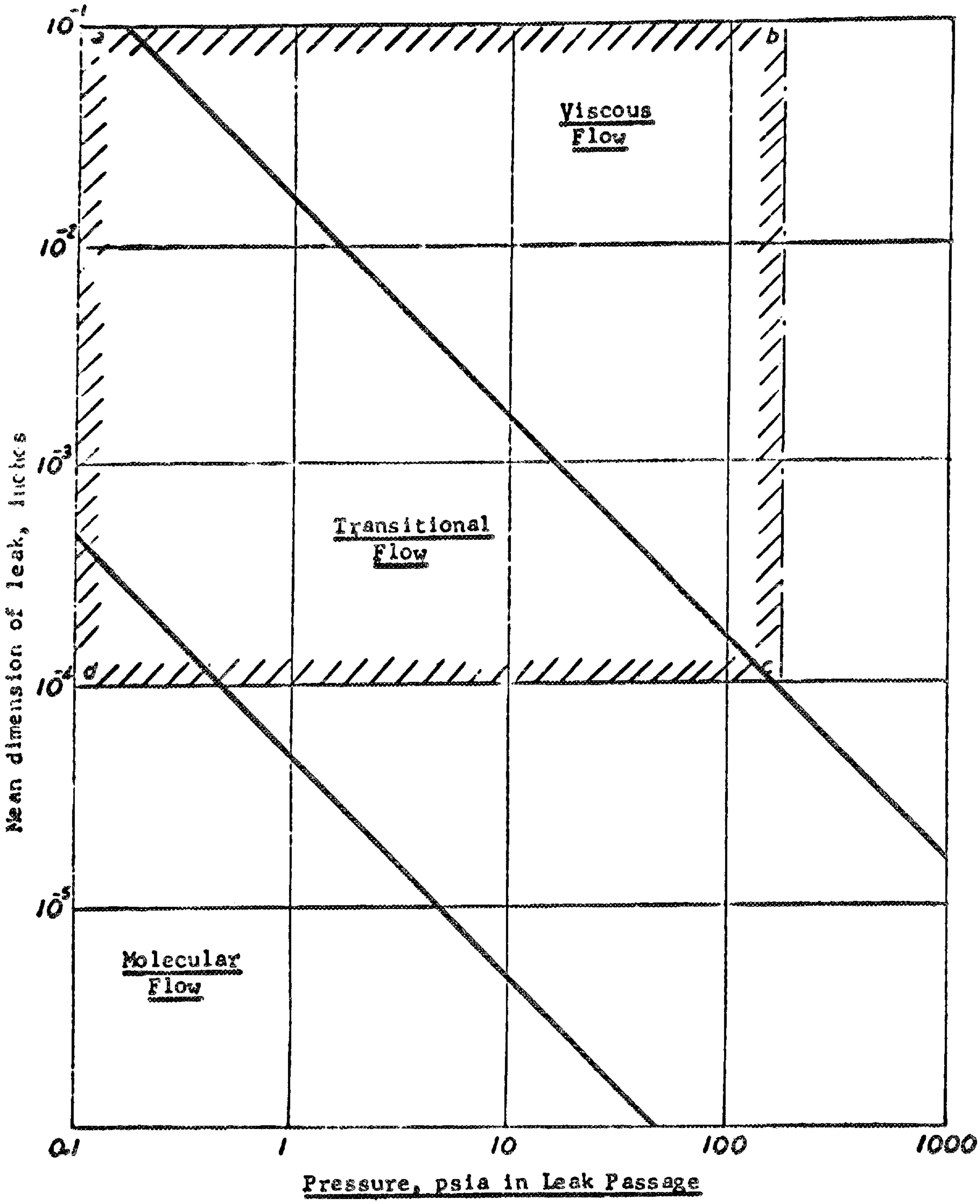


Fig. 10.1. Flow Regimes of Air at a Temperature of 70°F.

where $K_1'$ is the constant of proportionality; all other terms are defined in the nomenclature list above. Equation (10.3) may be further simplified by expressing the pressures in atmospheres (absolute*):

$$w = K_1'' \frac{D^3}{l} \frac{\overline{P}_a - 1}{(MRT)^{1/2}} . \tag{10.4}$$

No attempt is intended, in this instance, to present the numerous modifications applicable to leak geometries with other than circular cross sections because of the limited importance of molecular or transitional flows in containment leakage. It is sufficient to recognize that the mass rate of flow, w, for a given gas (air) is directly dependent upon the differential pressure across the leak ($P_a - P_o$) and the leak geometry factor $D^3/l$. Since the sizes of leaks in containment structures in which molecular flow is possible are exceedingly small, the magnitude of the flow rate, w, is, in consequence, also relatively minor.

2. Viscous (or Laminar) Flow. The general parameters that define the viscous flow of a gas through a straight leak passage of circular section are familiarly expressed by the Hagen-Poiseuille equation:[23,24]

$$P_a - P_o = \frac{8\mu V_s l}{r^2} . \tag{10.5}$$

The validity of this equation is predicated on the following assumptions implied in the development of the laminar flow law:

1. The gas is incompressible (density, ρ, constant).
2. The flow is laminar (no turbulent motion of the gas).
3. The flow velocity profile is constant throughout the length, $l$, of the leak passage.
4. The flow velocity at the leak walls is zero (wall friction forces are predominant).

For gas flowing through leaks at low velocities and with a low differential pressure across the leak, the flow may be considered as incompressible. However, with increased velocities, longer leak paths, and significant pressure differentials, the gas density is no longer constant between the entrance and the exit of the leak passage. The Hagen-Poiseuille equation must therefore be modified to treat the flow of a compressible fluid with its accompanying density changes. By substituting the following appropriate relations in Eq. (10.5),

---

*It must be recognized that Eq. (10.4) is based on a downstream absolute pressure of the leak path equal to atmospheric (barometric) pressure.

$$V_s = \frac{Q}{A} ,$$

$$A = \frac{\pi D^2}{4} ,$$

$$Q = \frac{w}{\rho_{av} g} ,$$

where $\rho_{av}$ represents the average density of the gas flowing through the leak as determined by the pressures $P_a$ and $P_o$, the laminar flow equation for a compressible fluid becomes

$$w = \frac{g \pi r^4}{8 \mu l} \rho_{av} (P_a - P_o) . \tag{10.6}$$

However, for the conditions at the leak, $\rho_{av}$ may be approximated (T constant) by

$$\rho_{av} = \frac{P_{av}}{RT} = \frac{P_a + P_o}{2} \frac{1}{RT} ,$$

and thus

$$w = \frac{g \pi D^4}{256 \mu l} \frac{1}{RT} (P_a^2 - P_o^2) . \tag{10.7}$$

Equation (10.7) is in the form commonly referred to as the Darcy formula. A typical application of this equation by Shapiro[25] is illustrated by a problem of laminar seepage of gas through a tube, which condition may be similar to leaks in containment vessels.

In terms of leak area, A, and a constant of proportionality, $K_2$, the equation becomes

$$w = K_2 \frac{AD^2}{\mu l} \frac{1}{RT} (P_a^2 - P_o^2) . \tag{10.8}$$

Grinnell[26] has further analyzed laminar flow of a compressible fluid in a thin, rectangular passage. This condition may be considered to apply to leak paths in containment structures such as at improperly sealed gasketed closures where a rectangular opening (narrow slit) is possible. The comparable equation for mass rate of flow, w, through a rectangular leak, as developed by Grinnell, is

$$w = K_2' \frac{Ah^2}{\mu l} \frac{1}{RT} (P_a^2 - P_o^2) \ , \qquad (10.9)$$

which indicates that the width of the crack (rectangular passage), h, is analogous to D for the diameter of a circular-section leak. The results of the experimental work of Grinnell further indicate that Eq. (10.9) is reasonably accurate for the lower values of $h/l$ that are within the range of interest in containment leaks.

To further simplify Eq. (10.8), the pressures $P_a$ and $P_o$ may be expressed in atmospheres:

$$w = K_2'' \frac{AD^2}{\mu l} \frac{1}{RT} (\overline{P}_a^2 - 1) \ . \qquad (10.10)$$

Although Eqs. (10.9) and (10.10) define the parameters that govern under laminar flow conditions, it should be recognized that these equations are applicable to simple leak geometries. In practice, the leak geometries are likely to deviate from the circular hole or the rectangular cross section. The laminar flow rate derived from the equations can therefore only be interpreted as a reasonable prediction of the leakage, which will be modified by the effect of the indeterminate leak geometries.

Since the region that defines the transition between laminar flow and turbulent flow cannot always be accurately established, the conservative practice is to assume laminar (viscous) flow for Reynolds numbers, $\rho V_s D/\mu$, below approximately 2000. Any variations from the assumed simple leak geometries that will readily influence the value of the Reynolds number can conceivably contribute to further variations in the flow rate.

The importance of the parameter $(P_a^2 - P_o^2)$ in laminar flow is evident from the significant effect that the pressure differential across the leak path has upon the mass flow rate, w. For a given pressure differential in a containment leak $(P_a - P_o)$, laminar flow conditions represent a noticeable increase of flow over molecular flow conditions. In addition, the leak flow area compatible with laminar flow development is generally much larger than that required for molecular flow. The contribution of laminar flow to containment leakage is undoubtedly of greater consequence than molecular flow.

Because of the various leak geometries (cross section dimensions and length of leak path) encountered in containment vessel structures, two leaks exhibiting the same flow rate at one pressure condition will probably not have the same flow rate for a different set of conditions. This effect, as reported by Santeler,[27] reveals that the difference in geometry in a porosity leak, a thin crack, or a round orifice causes a shift in the pressure range in which viscous flow may occur. Although all three leaks exhibited the same leakage at one particular pressure, the flow behavior was modified under different pressure conditions to other types of flow. This situation emphasizes the error that may be introduced in extrapolating containment vessel leakage-rate test results from a low pressure to a higher pressure.

3. Turbulent Flow. Although the majority of significant leaks in a containment vessel may well be in the laminar flow regime at one pressure,

it is conceivable that with higher velocities, turbulent flow conditions will be attained in irregular leak passages. The region of transition between viscous flow and turbulent flow may be conservatively estimated to begin with a Reynolds number, $N_R$, greater than 2000. Flow instabilities are usually present at the transition region between viscous and turbulent flow that make prediction of the type of prevailing flow uncertain. Transition effects as the gases enter and leave the irregularly shaped leak passage further complicate the flow character in the region of turbulent flow. If the leak path in a containment structure is of a nature to permit the development of turbulent flow, the mass flow rate, w, may be derived from the Darcy-Weisbach equation,[23,28] assuming that the leak cross section approaches the geometry of a small-diameter tube. The equation commonly appears as

$$P_a - P_o = f \frac{l\rho}{D} \frac{V^2}{2} , \tag{10.11}$$

where f is the friction factor, whose value is a function of the leak passage roughness and the Reynolds number for the flow conditions through the leak.

Since Eq. (10.11) is characterized for a circular leak of diameter D, another geometric parameter must be introduced for the cases of noncircular cross sections. In containment structures, leaks of noncircular cross section are more likely to exist than uniformly round leaks. By introducing the geometric parameter $R_H$, the hydraulic radius, the Darcy-Weisbach equation may be modified for noncircular leaks, as follows:

$$P_a - P_o = f \frac{l}{4R_H} \frac{\rho V^2}{2} . \tag{10.12}$$

The validity of Eqs. (10.11) and (10.12) is limited, however, to turbulent flow in leak passages of constant cross section for the entire length of the leak. Such conditions are not necessarily encountered with leaks in containment structures. The equations should, therefore, be interpreted as merely indicative of the parameters that govern in the turbulent flow regime for leak geometries which do not depart greatly from the generalized geometric parameters.

A gas flowing through a leak passage where $l/D$ and $\Delta P$ may be appreciable would not only encounter wall friction losses but it would also expand and increase its kinetic energy. Although under such conditions the behavior may change from incompressible to compressible flow, an approximation may be made by treating the problem as incompressible and assigning average values for the variables in Eq. (10.11). Thus,

$$P_a - P_o = f \frac{l}{D} \frac{\rho_{av} V_s^2}{2} , \tag{10.13}$$

where

$$\rho_{av} = \frac{P_a + P_o}{2} \frac{1}{RT(g)} ,$$

and

$$V_s = \frac{w}{P_{av} g A} ,$$

which, when substituted in Eq. (10.13), yield

$$w = g^{1/2} A \left[\frac{D}{fl} \frac{1}{RT} \left(P_a^2 - P_o^2\right)\right]^{1/2} , \qquad (10.14)$$

or, if pressures $P_a$ and $P_o$ are expressed in terms of atmospheres,*

$$w = K_3 A \left[\frac{D}{fl} \frac{1}{RT} \left(\overline{P}^2 - 1\right)\right]^{1/2} . \qquad (10.15)$$

Knapp and Metzgar[29] have developed the mass flow rate equation in a form similar to Eq. (10.13) from a more rigorous treatment of the differential equation of one-dimensional compressible flow in conjunction with frictional losses of air and steam flowing turbulently at low pressure. Because of the indeterminate nature of the leak surface roughness, any attempt to assign a value to the friction factor in Eq. (10.14) can only be interpreted as an approximate estimate. Instead, for flows with Reynolds numbers $4 \times 10^3$ to $10^5$, it may be assumed that the profile of flow through the leak approaches that of a smooth tube for which the empirical expression of Blasius law[30] may be applied in the determination of a conservative friction factor. For circular leaks,

$$f = \frac{0.316}{N_R^{1/4}} = \frac{0.316}{\left(\frac{VD\rho}{\mu}\right)^{1/4}} , \qquad (10.16)$$

and for noncircular leaks,

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*Again, it is to be noted that the downstream pressure $P_o$ is assumed as atmospheric (barometric) pressure.

$$f = \frac{0.316}{\left(\frac{4VR_{H}\rho}{\mu}\right)^{1/4}} . \tag{10.16a}$$

Substituting Eq. (10.16) in the Darcy-Weisbach equation (10.14) and rearranging terms yields

$$w = K_4 A \frac{D^{5/7}}{\mu^{1/7}} \frac{1}{(lRT)^{4/7}} \left(P_a^2 - P_o^2\right)^{4/7} , \tag{10.17}$$

which may also be expressed in terms of $P_a$ and $P_o$ in atmospheres,* as

$$w = K_4' A \frac{D^{5/7}}{\mu^{1/7}} \left(\frac{\overline{P}_a^2 - 1}{lRT}\right)^{4/7} . \tag{10.18}$$

Although this approach tends to generalize the actual conditions of flow through leaks that may be better categorized as "rough" tubes, the results obtained in mass flow rate from Eq. (10.18) will yield values on the high side, except possibly for usually unstable flow in the transition region between viscous and turbulent flow.

The relationships defined by Eqs. (10.15) and (10.18) should be recognized as indicative of the influence of pressure, temperature, and viscosity on turbulent flow through an idealized leak of circular cross section. It is inconceivable that many leaks existing in containment vessels would have smooth, straight surfaces and a uniform circular cross section for the entire length of the flow path. Instead, it would be more likely that the leak would have an irregular surface and a variable flow area. Notwithstanding, the above equation serves to define, in a general way, the more important parameters affecting flow in the turbulent domain for the purpose of a comparative study of leak flow behavior.

4. Compressible Flow Through Convergent Passages or Orifices. An important category of leak that may exist in a containment structure may be characterized by the geometry normally associated with nozzles and orifices. Leaks with convergent passages may be considered as nozzles, while leaks whose length is relatively short compared with the diameter may readily behave as orifices. Significant changes in gas density occur with changes in velocity in the case of compressible flow through nozzles and orifices that involve the thermodynamic effects of the compressibility of the gas. The flow behavior, then, depends on the velocity attained in the gas stream through the leak, and the corresponding sonic velocity of the gas at the flow conditions.

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*The downstream pressure is assumed to be atmospheric (barometric) pressure as in previously derived equations.

For air, the sonic velocity is given by the relation,

$$c = 49.02T^{1/2} . \qquad (10.19)$$

In the case of dry air, the sonic velocity, c, at 70°F is 1130 ft/sec. It is to be noted that the containment atmosphere during leakage-rate tests is never entirely dry air and that in the event of a loss-of-water-coolant accident, the atmosphere is essentially a steam-air mixture. In such cases, the sonic velocity will be modified both by the temperature and the composition of containment atmosphere. The value of the Mach number, $N_m$, that determines the effect of downstream pressure on flow will be correspondingly modified.

The Mach number, $N_m$, is a convenient parameter, which is defined as the ratio of velocity of gas flow, $V_c$, to the sonic velocity, c, and serves to distinguish between the physical significance of subsonic and supersonic flow velocities. When a compressible flow is at subsonic velocity, $N_m < 1$, the flow through a leak is always influenced by the downstream pressure conditions. On the other hand, at velocities $N_m > 1$, the downstream pressure conditions cannot affect the leakage, assuming orifice geometries exist.

Obviously the upstream and downstream pressures at a leak will influence its mass rate of flow. These pressures in a containment structure are, respectively, the inside containment pressure, $P_a$, and the outside ambient atmospheric pressure, $P_o$. At flows corresponding to $N_m = 1$ (sonic flow), the ratio $P_o/P_a$ becomes the critical pressure ratio, which, for air at normal conditions (with $\kappa = 1.4$), attains a value of 0.528. (For saturated steam, the critical pressure ratio is 0.578, which gives an indication that an air-steam mixture would have an intermediate value.) In a containment vessel, the pressure conditions that can exist (during testing with air) may be classified in relation to this critical pressure ratio for 2 cases:

Case 1. For containment vessel pressures equal to or below 27.8 psia (critical pressure ratio, 14.7/0.528 = 27.8), the flow velocity through leaks will be subsonic, $N_m < 1$.

Case 2. For containment vessel pressures greater than 27.8 psia (subcritical pressure ratios), the flow velocity through leaks will be sonic, $N_m = 1$.

The variation of the mass flow rate, w, through an orifice-type leak can then be determined by applying the familiar compressible flow equation for one-dimensional adiabatic flow:[30]

Case 1. For $P_o/P_a \geq 0.528$ and $P_a \leq 27.8$ psia, when $P_o = 14.7$ psia,

$$w = C_d A \left\{ \frac{P_a^2}{RT} \frac{2\kappa}{\kappa - 1} g \left[ 1 - \left( \frac{P_o}{P_a} \right)^{(\kappa-1)/\kappa} \right] \right\}^{1/2} \left( \frac{P_o}{P_a} \right)^{1/\kappa} . \qquad (10.20)$$

Case 2. For $P_o/P_a < 0.528$ and $P_a > 27.8$ psia, when $P_o = 14.7$ psia,

$$w = C_d A \left( \frac{P_a^2}{RT} \frac{2\kappa}{\kappa + 1} g \right)^{1/2} \left( \frac{2}{\kappa + 1} \right)^{1/(\kappa-1)} . \qquad (10.21)$$

Thus, any leakage that is below sonic velocity (containment pressure less than 27.8 psia) is dependent upon both upstream ($P_a$) and downstream ($P_o$) pressures, while leakage that attains sonic velocity (containment pressure greater than 27.8 psia) is independent of the downstream pressure ($P_o$). Equations (10.20) and (10.21) may be expressed in the analogous manner indicated for other flow equations, that is, in terms of atmospheres and a constant of proportionality. For case 1,

$$w = K_5 A \left[ \frac{1}{RT} \frac{2\kappa}{\kappa - 1} \left( 1 - \frac{1}{\bar{P}_a^{(\kappa-1)/\kappa}} \right) \right]^{1/2} \bar{P}_a^{(\kappa-1)/\kappa} c_p , \qquad (10.22)$$

and for case 2,

$$w = K_5 A \left( \frac{1}{RT} \frac{2\kappa}{\kappa + 1} \right)^{1/2} \left( \frac{2}{\kappa + 1} \right)^{1/(\kappa-1)} \bar{P}_a c_v . \qquad (10.23)$$

Since the orifice leak geometry is indeterminate for the majority of such leaks in containment structures, the value of the coefficient of discharge $C_d$ can only be estimated. The experimental work of Grace and Lapple[31] presents results which indicate the variation of the coefficient of discharge of defined orifice geometry for different ratios of downstream to upstream pressures. The range of this variation is approximately 0.55 to 0.99; it is dependent upon the specific orifice geometry. For an assumed orifice flow behavior of leaks in a containment structure, the discharge coefficient that is applicable is, at best, only a guess because of the many undefined orifice leak geometries possible in such structures.

### 10.4.5.2 Predominant Leakage Characteristics

To provide an indication of the relative magnitude of flow of air through containment leaks, as governed by the relationships of Eqs. (10.4), (10.10), (10.15), (10.18), (10.22), and (10.23) for the flow regimes investigated, a hypothetical example was selected to prepare Fig. 10.2. In this example, it was assumed that a containment structure contained a number of leaks whose total flow area was equivalent to that of an orifice having a diameter of 0.12 in. (0.01 ft). The total flow, w, through the leaks was then calculated on the basis of the pounds of air escaping per day from the containment vessel as a function of the containment pressure. It was further assumed that all leaks possessed the geometry required to ensure the same flow behavior. The calculated size and number of leaks were different for each pressure to satisfy the governing parameters in

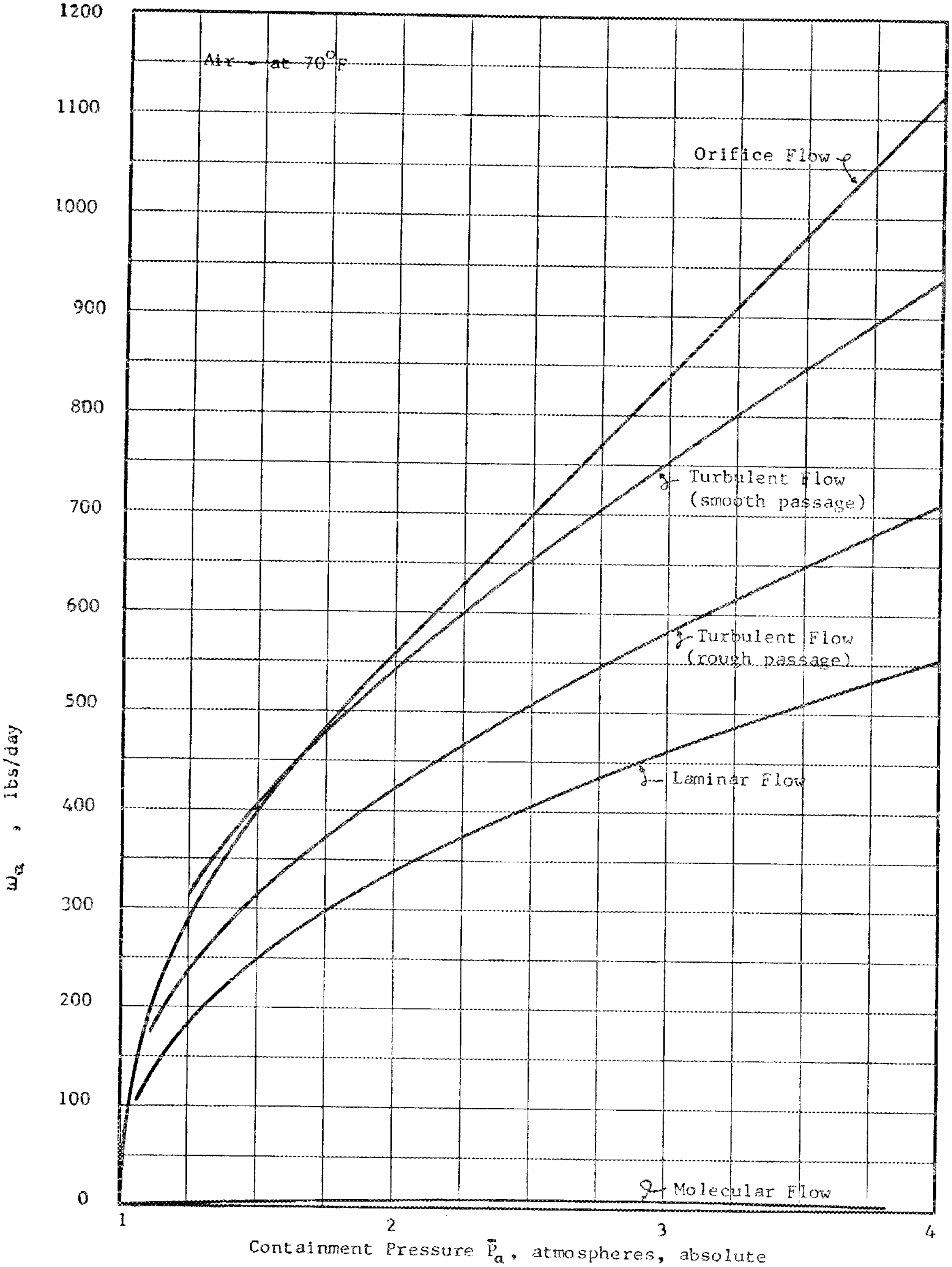


Fig. 10.2. Comparative Flow Through Leaks Based on Assumption of Equal Flow Area at a Selected Containment Pressure.

each flow regime. Figure 10.2 clearly shows that leaks which behave as orifices will allow the largest flow of air when comparison with other flows is made on the basis of equal areas. In practice, however, the assumptions of the example are not likely to exist in containment structures.

With the knowledge of the parameters that control the leakage in a containment structure, as derived from the relationship in each flow regime investigated, it would then be desirable to learn whether containment leakages manifest any predominant leakage characteristic. Only limited experience and leakage tests are available from which justifiable and reliable deductions can be made.

Although for many reported containment leakage tests it has been assumed that compressible flow through orifices is representative of the leakage behavior, no available test data or evidence justify the assumption or support this conclusion. Each containment structure should preferably be considered as uniquely influenced by several or all of the following factors that ultimately determine its leakage rate:

1. The pressure and temperature conditions of the containment atmosphere,
2. The composition and physical properties of the containment atmosphere (in the case of accident conditions),
3. The number of potential leakage paths,
4. The dissimilarity of flow characteristics among the existing leaks,
5. The variation of leak geometry with variations of internal containment pressures,
6. The extent of the containment boundary limits beyond the containment shell.

From a practical standpoint, the containment total leakage should therefore be recognized as merely representative of a composite average of the above-mentioned effects, and no attempt should be made to analyze the leakage dependence of the individual contributing factors. If containment leakage tests indicate characteristics whose combined effects may approximate any of the flow regimes, the equations presented in this section may then be employed to analyze the expected leakage behavior at different test pressures. In some cases, no predominant characteristic may be discernible. In any case, conservative judgment should be exercised in weighing the validity of extrapolated leakage results.

### 10.4.5.3 Leak Locations

Test experience has shown that the most likely locations of leaks in a containment system that has been carefully built and tested are the access openings and the ventilation system isolation valves. Because these parts may be opened and closed repeatedly, they are more likely to be subject to improper sealing due to malfunction or deterioration of sealing surfaces. Therefore, the air locks, gasketed doors and hatches, and ventilation system valves are the first places to check for leakage, both in initial leakage tests and in retests. It has been proposed (see Sec. 10.4.10.3) that once an integrated leakage-rate test has demonstrated adequate containment vessel leaktightness, subsequent leakage tests be conducted locally only on areas where seal deterioration is most likely provided these leaks have been initially characterized and related to total containment leakage. Leakage tests of the vapor container at Shippingport[32,33] after considerable plant operation showed that the ventilation system butterfly valves and the air locks had experienced large increases in leakage. Repairs and adjustments were required to achieve allowable leakage rates.

Penetrations of the containment vessel wall for piping and for electrical and instrumentation wiring are also major areas for concern as potentially significant leak paths. These penetrations must be carefully designed and fabricated but, because of the large number of penetrations that may exist in a containment vessel, there is a reasonable probability that significant leakage may occur in at least one penetration. However, initial local leak testing should locate any such leaks. Thermal and pressure cycling, if severe, could cause some leaks, so penetrations or other areas subject to such conditions should be inspected periodically. Various penetration designs have been used with success in existing plants. Some of these designs are described in Chapter 9.

Leakage through welds is quite unlikely in view of the careful inspection required of all vessel welds. Any porosity or other flaws in a weld that might permit significant leakage would be detected during normal weld inspection or during the initial soap bubble test and would cause the weld to be rejected and repaired.

### 10.4.6 Variation of Leakage Rate with Pressure

Extrapolating a leakage rate measured at one pressure to an equivalent leakage rate at another pressure is of particular interest for leakage rate testing of containment vessels. It may be difficult or impractical to perform a leakage rate test at full design pressure, particularly in the case of retesting the vessel after all equipment has been installed (see Sec. 10.6). It is then often necessary to conduct the test at a reduced pressure and to extrapolate the result to the design pressure for comparison with the specified maximum allowable leakage rate. The use of results from low-pressure tests and extrapolation to accident pressures should preferably be avoided, particularly in those containment systems in which operation of engineered safeguards is depended upon and the related instrumentation must be functional in the event of an accident. In certain special cases, it may be desirable to conduct an accelerated leakage-rate test at a pressure higher than design pressure (see Sec. 10.4.13.2) and to extrapolate the measured leakage rate to the lower pressure (or to extrapolate the specified rate to an equivalent rate at the test pressure). It also may be important to know how much the leakage rate will decrease when the containment pressure decreases below the peak pressure reached during an accident. In each of these cases it is necessary to know the relationship between pressure and leakage rate over a fairly wide range of pressure for the types of leaks that might exist in containment vessels.

As was indicated in the previous section, the flow rate through a given opening will normally increase in proportion to some power of the differential pressure. However, extrapolating a containment vessel leakage rate determined at one pressure to a realistic rate at another pressure poses several problems. First, there is no assurance that the type, number, and size of leaks will be the same at the two pressures. Even disregarding expansion of openings with pressure, there may be some very significant leaks that are not evident until a minimum pressure is exceeded. On the other hand, seals and gaskets may pressure seal so that an increase in pressure improves the seal and reduces leakage. Also, since the exact configurations of the leakage paths are not known and since the type of flow may change as the pressure is changed, the type of flow through the leaks and, hence, the way in which flow rate varies with pressure are not known.

In view of the multiplicity and varying geometry of leaks existing in a structure, any attempt to distinguish between the various types of flow inherently combined in the leakage-rate tests is futile. The only recourse, in such cases, is either to assume a leak flow behavior that yields conservative leakage rates or to perform a series of tests that establish the leakage behavior.

If a majority of the leaks in the containment vessel are of the character to respond in accordance with a predominant flow, the characteristics of this flow behavior over a range of test pressures may be recognized in the analysis of leak test data by comparing the measured leakage rate with the calculated leakage rate expected under the various types of flow investigated (see Sec. 10.4.5). A conservative correction factor may then be applied to the test results to approximate the leakage under accident conditions. The results thus obtained can only be viewed as a conservative approximation, because the tests cannot duplicate entirely the conditions expected in the event of an accident.

An an example to illustrate this point, the passage through very small leaks of a steam-air mixture may result in a fluid change of state that can alter the flow characteristics. Moisture plugging in small leak passages, which can conceivably contribute to an appreciable reduction in leakage, particularly at lower pressures, has not been adequately investigated. On the other hand, the discharge of condensed moisture containing radioactive materials may offset the leakage reduction because of the containment pressure tending to expel water accumulation in larger leak passages.

### 10.4.6.1 Leakage-Rate Definition

In containment structures, the leakage, L, of the system is generally expressed as a ratio of the change in weight of the containment atmosphere, $W_o - W_1$, between an initial (o) and a final (1) state, to the initial total weight $W_o$ of the free volume of the containment, or

$$L = \frac{W_o - W_1}{W_o} . \tag{10.24}$$

The leakage thus expressed is a fractional number and should not be confused with any measure of the weight quantity of flow (although it serves to calculate this quantity). The leakage, L, represents, therefore, the fractional loss of the containment atmosphere, and, when it is projected on a time basis, it is the leakage rate, $L_r$, of the system:

$$L_r = \frac{\dfrac{W_o - W_1}{W_o}}{\Delta t} . \qquad (10.25)$$

The most commonly selected time interval, $\Delta t$, is 24 hr, in which case the leakage rate, $L_r$, becomes the fractional weight loss of the initially contained atmosphere over a 24-hr period. Since the containment atmosphere weight difference $(W_o - W_1)/\Delta t$ is the weight loss per interval of time, or the flow rate of escaping gas, w (as derived in the equation for the different modes of flow in Sec. 10.4.5), the leakage rate can also be expressed as

$$L_r = \frac{w}{W_o} , \qquad (10.26)$$

where w is rate of flow through leaks in 24 hr and may be calculated from any of the applicable equations of Section 10.4.5. The initial weight of contained atmosphere, $W_o$, is determined from the equation of state for a perfect gas, assuming uniform conditions throughout the containment volume (not necessarily valid for noncirculated atmospheres):

$$W_o = \frac{P_a V}{RT} . \qquad (10.27)$$

By substitution, the leakage rate, $L_r$, is then conveniently expressed in terms of defined parameters:

$$L_r = w \frac{RT}{P_a V} . \qquad (10.28)$$

10.4.6.2 Extrapolation of Leakage Rates

Containment leakage rates have been determined in practice by conducting, in the majority of installations, pneumatic tests at a pressure considerably lower than the design pressure of the containment system. The values obtained from these low-pressure tests are then extrapolated to the desired higher pressure on the basis of unverified assumptions of the leak flow characteristics of the containment structure within its range of pressures.

In recognition of the problems, factors have been derived, as discussed below, that may be applied for each of the leak flow modes investigated in Section 10.4.5 in extrapolating the low-pressure test results to other pressure conditions. The extrapolation factors have been developed for application to two conditions:

1. Case 1, extrapolation from reduced-pressure air test conditions to a design-pressure air condition at the same temperature as the test,
2. Case 2, extrapolation from reduced-pressure air test condition to accident conditions.

The extrapolation factors for case 1 are denoted by subscripts t and e, while for case 2 the subscripts a and b are used. Traube[34] has developed similar extrapolation factors for condition 2 in an endeavor to justify low-pressure tests of containment vessels, but the conclusions drawn regarding these factors are in disagreement with available containment test data.

From the analysis of the extrapolation factors, it is considered possible to select factors that provide reasonable conservatism in extrapolated leakage rates. However, such procedures should not be construed as a substitute for reliable containment leakage tests at full design pressure until much more test data become available in support of extrapolation practices.

1. Definition of Leakage Extrapolation Factors. The extrapolation factor may be conveniently expressed as a dimensionless ratio of the leakage rates for the two cases of interest: for case 1, $L_e/L_t$, and for case 2, $L_b/L_a$. From Eq. (10.28), the case 1 factor is converted in terms of calculable parameters:

$$\frac{L_e}{L_t} = \frac{w_e \dfrac{RT}{P_e V}}{w_t \dfrac{RT}{P_t V}} . \tag{10.29}$$

Since V, R, and T are considered as constant, both for test conditions and extrapolated conditions, the extrapolation factor becomes

$$\frac{L_e}{L_t} = \frac{w_e}{w_t}\frac{P_t}{P_e} . \tag{10.30}$$

Similarly, the extrapolation factor for case 2 may be analogously expressed, but, in this instance, only the containment free volume, V, is considered constant:

$$\frac{L_b}{L_a} = \frac{w_b}{w_a} \frac{P_a}{P_a} \frac{R_b T_b}{R_a T_a} . \tag{10.31}$$

2. Leakage Rate as a Function of Pressure. Before deriving the relationships of the extrapolation factors, it is of interest to examine the role that containment pressure plays in influencing the leakage rate. To depict the dependence of leakage rate upon the absolute (air) pressure in the containment vessel, Figs. 10.3 through 10.7 show the relationships among the various types of flow through leaks. In the preparation of the graphs, the mass flow rate equations (Sec. 10.4.5) were made tractable by combining all factors, except pressure, into a single constant K and then substituting the resultant equations into Eq. (10.28):

Molecular Flow

$$L_r = K_m \left(1 - \frac{1}{\overline{P}_a}\right) . \tag{10.32}$$

Laminar (Viscous) Flow

$$L_r = K_l \left(\overline{P}_a - \frac{1}{\overline{P}_a}\right) . \tag{10.33}$$

Turbulent Flow, Rough Passage

$$L_r = K_{tr} \left(1 - \frac{1}{\overline{P}_a^2}\right)^{1/2} . \tag{10.34}$$

Turbulent Flow, Smooth Passage

$$L_r = K_{ts} \left(1 - \frac{1}{\overline{P}_a^2}\right)^{4/7} \overline{P}^{1/7} . \tag{10.35}$$

Compressible Flow Through Orifice Leaks

For $\overline{P}_a < 1.9$ atm

$$L_r = K_o \frac{1}{\overline{P}_a^{0.715}} \left(1 - \frac{1}{\overline{P}_a^{0.286}}\right)^{1/2} . \tag{10.36}$$

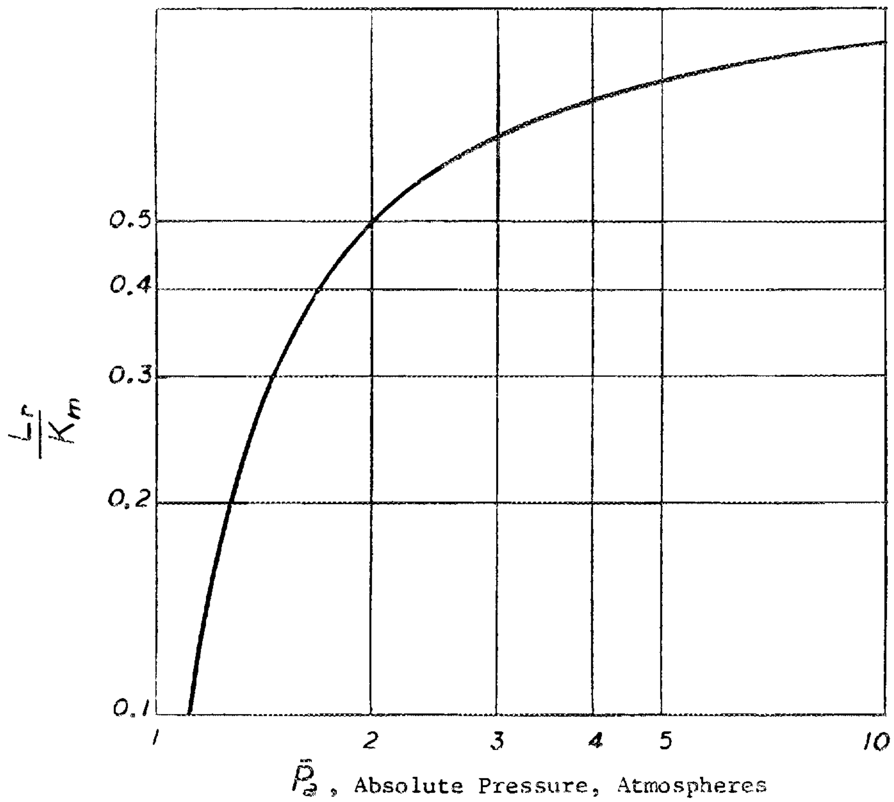


Fig. 10.3. Pressure Relationship for Molecular Flow.

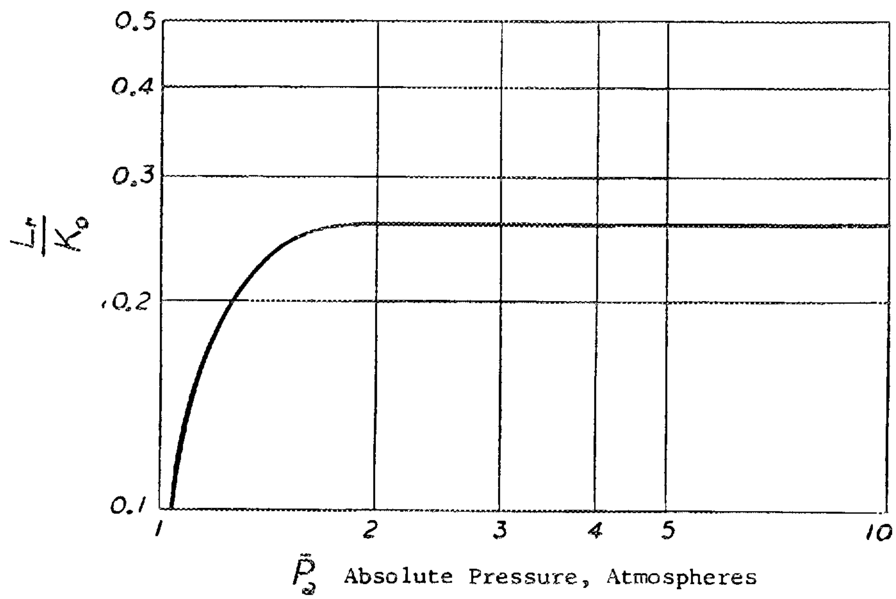


Fig. 10.4. Pressure Relationship for Orifice Flow.

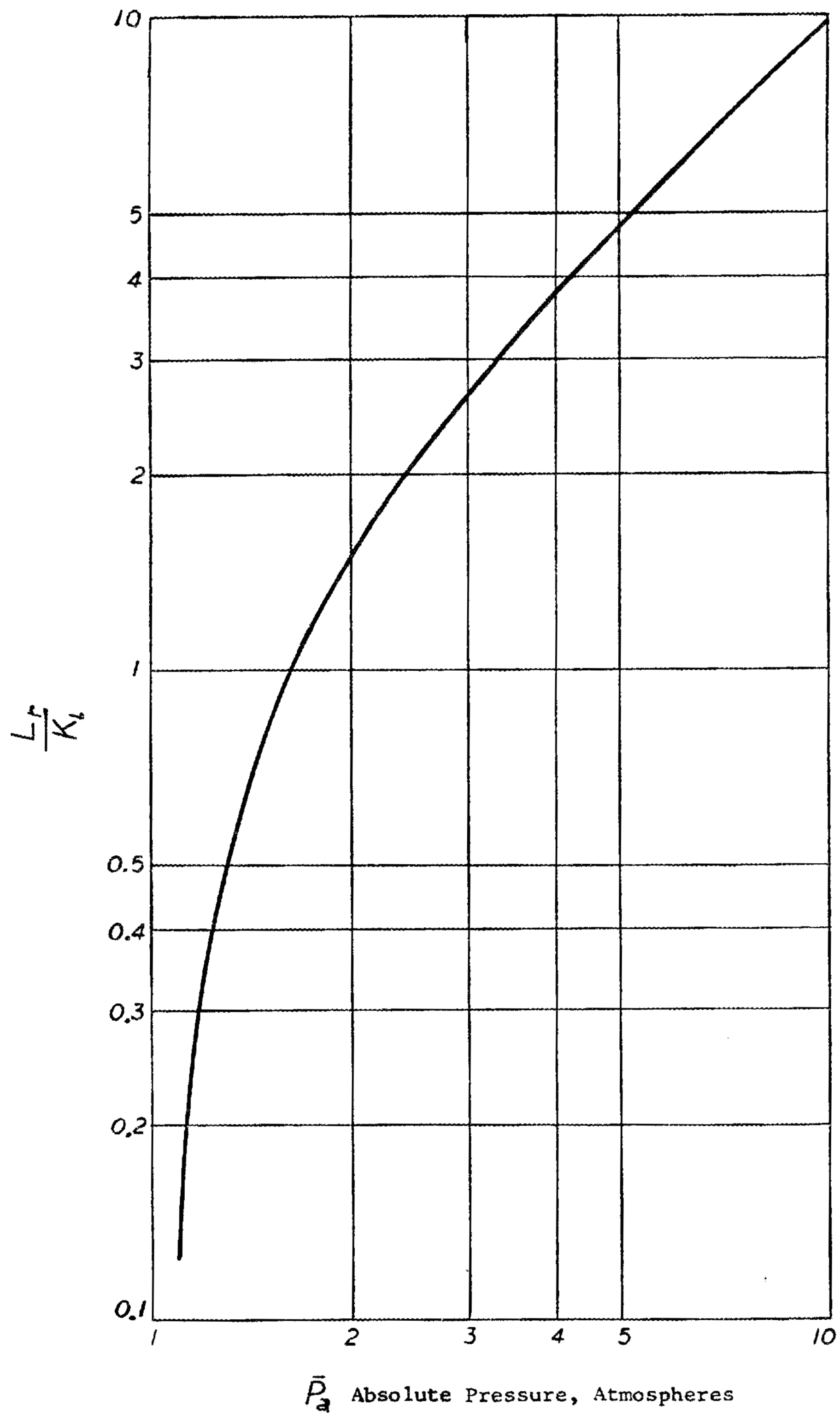


Fig. 10.5. Pressure Relationship for Laminar (Viscous) Flow.

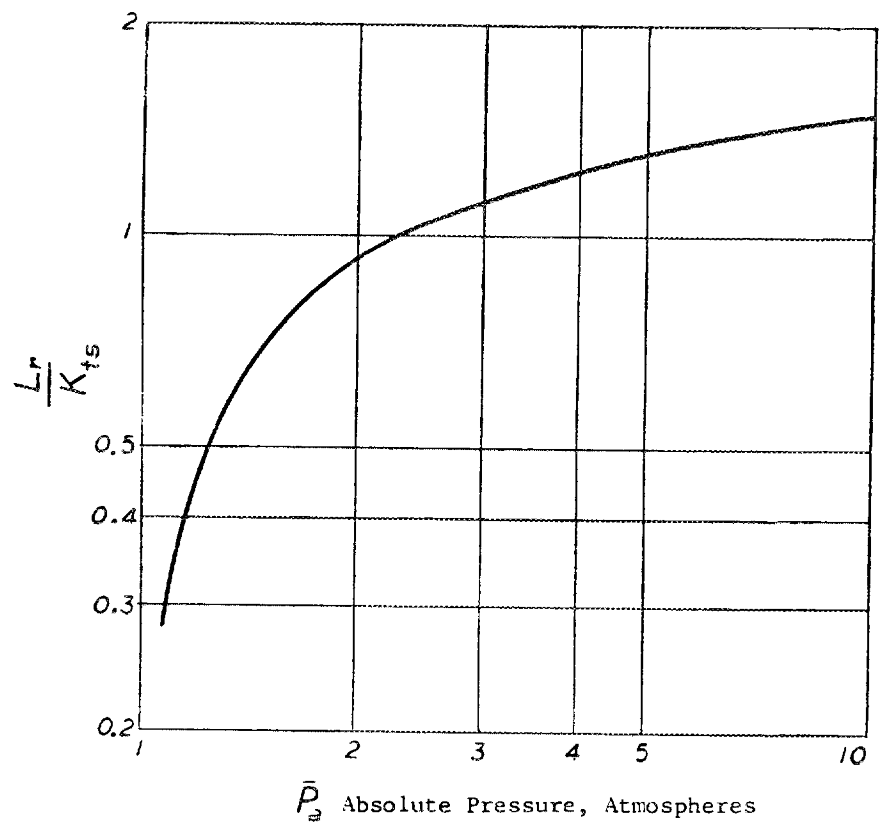


Fig. 10.6. Pressure Relationship for Turbulent (Smooth Passage) Flow.

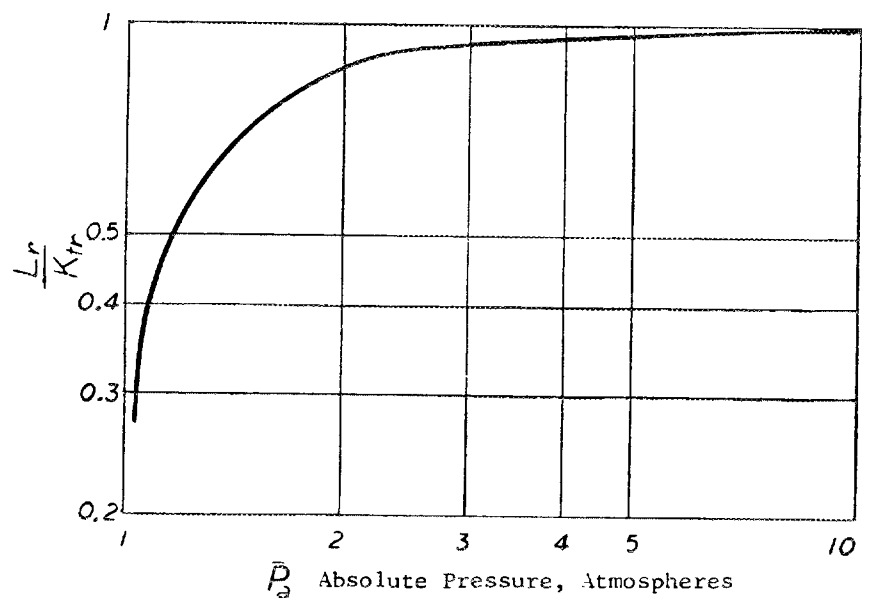


Fig. 10.7. Pressure Relationship for Turbulent (Rough Passage) Flow.

For $\overline{P}_a \geq 1.9$ atm

$$L_r = 0.259\ K_o\ . \tag{10.37}$$

Since the constants $K_m$, $K_l$, $K_{tr}$, $K_{ts}$, and $K_o$ are not equivalent, the influence of pressure, as shown in Figs. 10.3 through 10.7, is not to be interpreted as the only factor governing leakage rate. The constant K includes, of necessity, the factor n, number of leaks, which is generally unknown but can markedly change the total leakage rate. It also includes leak geometry.

From Fig. 10.5, laminar flow manifests the largest variation with pressure, while, from Fig. 10.4, orifice flow in the sonic velocity range is independent of the pressure. All flow rates through leaks are, however, directly dependent on the leak flow area. The existence of even a single orifice-type leak (partially open isolation valve) may produce a high leakage rate that will make the pressure dependence of other types of flow meaningless.

3. Extrapolation Factors. By substituting, in Eq. (10.30), the respective relationship expressed by w for each mode of flow investigated under Section 10.4.5, the corresponding extrapolation factors may be derived for case 1. Similarly, substitutions in Eq. (10.31) yield the factors for case 2.

The extrapolation factors then serve to define the parameters that influence the measured leakage rate when extrapolated to either a higher air pressure than test air pressure (case 1) or to an accident condition (case 2) where the properties of the containment atmosphere differ substantially from the conditions under which the leakage rate was measured.

All the factors are derived on the basis of four simplifying assumptions:

1. All leaks in containment systems behave in accordance with the single mode of flow for which the extrapolation factor is applicable.

2. All leak geometries (area of flow, leak path length, roughness of passage, etc.) remain constant for the pressure and temperature range of interest.

3. The number of leaks in the containment system does not vary with pressure or temperature and is not influenced by the composition of the containment atmosphere (no plugging).

4. The conditions within the containment system remain constant for the time interval of interest, as defined by the leakage rate (usually 24 hr).

It is recognized that all these assumptions are not necessarily justifiable for any particular containment system. The extrapolation factors are, therefore, to be interpreted as merely indicative of the relationship among the controlling parameters from which conservative values may be derived. By comparing the various extrapolation factors, it is possible to recognize those factors that yield conservative results. Application of the extrapolation factors in analyzing leakage-test data provides a means of estimating the predominant mode of flow through the leaks and the prediction of leakage characteristics of the containment system for other than the test conditions.

Case 1. All extrapolation factors are expressed in terms of atmospheres absolute to obtain the simplest form.

a. Molecular Flow (reference, Eq. 10.4)

$$\frac{1 - \dfrac{1}{\overline{P}_e}}{1 - \dfrac{1}{\overline{P}_t}} \; . \tag{10.38}$$

b. Laminar (Viscous) Flow (reference, Eq. 10.10)

$$\frac{L_e}{L_t} = \frac{\overline{P}_e - \dfrac{1}{\overline{P}_e}}{\overline{P}_t - \dfrac{1}{\overline{P}_t}} \; . \tag{10.39}$$

c. Turbulent Flow, Rough Leak Passage and Constant Friction Factor (reference, Eq. 10.15)

$$\frac{L_e}{L_t} = \left( \frac{1 - \dfrac{1}{\overline{P}_e^2}}{1 - \dfrac{1}{\overline{P}_t^2}} \right)^{1/2} \; . \tag{10.40}$$

d. Turbulent Flow, Smooth Leak Passage and Variable Friction Factor (reference, Eq. 10.18)

$$\frac{L_e}{L_t} = \left( \frac{\overline{P}_e}{\overline{P}_t} \right)^{1/7} \left( \frac{1 - \dfrac{1}{\overline{P}_e^2}}{1 - \dfrac{1}{\overline{P}_t^2}} \right)^{4/7} \; . \tag{10.41}$$

e. Compressible Air Flow Through Orifices (reference, Eqs. 10.22 and 10.23)

For the conditions, $\overline{P}_e, \overline{P}_t < 1.9$ atm,

$$\frac{L_e}{L_t} = \left(\frac{\bar{P}_t}{\bar{P}_e}\right)^{0.715} \left(\frac{1 - \dfrac{1}{\bar{P}_e^{0.286}}}{1 - \dfrac{1}{\bar{P}_t^{0.286}}}\right)^{1/2} \frac{C_{de}}{C_{dt}} \ . \tag{10.42}$$

For the conditions, $\bar{P}_e > 1.9$ atm and $\bar{P}_t < 1.9$ atm,

$$\frac{L_e}{L_t} = \frac{0.259}{\dfrac{1}{\bar{P}_t^{0.715}} \left(1 - \dfrac{1}{\bar{P}_t^{0.286}}\right)^{1/2}} \frac{C_{de}}{C_{dt}} \ . \tag{10.43}$$

For the condition, $\bar{P}_e, \bar{P}_t > 1.9$ atm,

$$\frac{L_e}{L_t} = \frac{C_{de}}{C_{dt}} \ . \tag{10.44}$$

Case 2

a. Molecular Flow

$$\frac{L_b}{L_a} = \frac{1 - \dfrac{1}{\bar{P}_b}}{1 - \dfrac{1}{\bar{P}_a}} \frac{R_b}{R_a} \left(\frac{T_b}{T_a}\right)^{1/2} \ . \tag{10.45}$$

b. Laminar (Viscous) Flow

$$\frac{L_b}{L_a} = \frac{\bar{P}_b - \dfrac{1}{\bar{P}_b}}{\bar{P}_a - \dfrac{1}{\bar{P}_a}} \frac{\mu_a}{\mu_b} \ . \tag{10.46}$$

c. Turbulent Flow, Rough Leak Passage and Constant Friction Factor

$$\frac{L_b}{L_a} = \left( \frac{1 - \frac{1}{\overline{P}_b^2}}{1 - \frac{1}{\overline{P}_a^2}} \frac{R_b T_b}{R_a T_a} \right)^{1/2} . \qquad (10.47)$$

d. Turbulent Flow, Smooth Leak Passage and Variable Friction Factor

$$\frac{L_b}{L_a} = \left( \frac{\overline{P}_b}{\overline{P}_a} \frac{\mu_a}{\mu_b} \right)^{1/7} \left( \frac{1 - \frac{1}{\overline{P}_b^2}}{1 - \frac{1}{\overline{P}_a^2}} \right)^{4/7} \left( \frac{R_b T_b}{R_a T_a} \right)^{3/7} . \qquad (10.48)$$

e. Compressible Flow Through Orifice

For the condition, $P_b, P_a < 1.9$ atm,

$$\frac{L_b}{L_a} = \left[ \frac{R_b T_b}{R_a T_a} \frac{\kappa_b(\kappa_a - 1)}{\kappa_a(\kappa_b - 1)} \frac{1 - \frac{1}{\overline{P}_b^{(\kappa_b - 1)/\kappa_b}}}{1 - \frac{1}{\overline{P}_a^{(\kappa_a - 1)/\kappa_a}}} \right]^{1/2} \frac{\overline{P}_a^{1/\kappa_a}}{\overline{P}_b^{1/\kappa_b}} \frac{C_{db}}{C_{da}} . \qquad (10.49)$$

For the condition, $\overline{P}_b > 1.9 \text{ atm} > \overline{P}_a$,

$$\frac{L_b}{L_a} = \left[ \frac{\frac{R_b T_b}{R_a T_a} \frac{\kappa_b(\kappa_a - 1)}{\kappa_a(\kappa_b + 1)}}{1 - \frac{1}{\overline{P}_a^{(\kappa_a - 1)/\kappa_a}}} \right]^{1/2} \frac{\left( \frac{2}{\kappa_b + 1} \right)^{1/(\kappa_b - 1)}}{\left( \frac{1}{\overline{P}_a} \right)^{1/\kappa_a}} \frac{C_{db}}{C_{da}} . \qquad (10.50)$$

For the condition, $\overline{P}_b, \overline{P}_a > 1.9$ atm,

$$\frac{L_b}{L_a} = \left[ \frac{\kappa_b}{\kappa_a} \frac{(\kappa_a + 1)^{(\kappa_a + 1)/(\kappa_a - 1)}}{(\kappa_b + 1)^{(\kappa_b + 1)/(\kappa_b - 1)}} \frac{R_b T_b}{R_a T_a} \right]^{1/2} \left[ 2^{(\kappa_a - \kappa_b)/(\kappa_b - 1)(\kappa_a - 1)} \right] \frac{C_{db}}{C_{da}} . \qquad (10.51)$$

For an air-steam mixture in the containment vessel, the critical pressure is slightly less than 1.9 atmospheres and depends upon the air-to-steam ratio. Equations (10.38) through (10.44) have been plotted in Fig. 10.8 for the specific condition* in which the test pressure is 10 psig. From this figure the extrapolation factor may be read directly for any ratio of $\overline{P}_e/\overline{P}_t$. Similar graphs may be prepared for any other test pressure.

A study of Fig. 10.8 shows that for extrapolation of leakage rates from a low test pressure (10 psig) to a higher pressure, laminar flow will yield a larger leakage rate than other modes of flow. In the absence of leakage-test data at the higher pressure, it appears that for upward extrapolation the laminar flow can be considered as most conservative, while extrapolation based on orifice flow will provide the least conservative value for the leakage rate.

To extrapolate downward (from a high pressure to a 10 psig pressure), the assumption of orifice flow will obviously result in the least change in the leakage-rate value, since the extrapolation factor is 1.0 (ratio $C_{de}/C_{dt}$ is assumed equal to 1) for the entire range of pressure ratios for which sonic velocity exists in the orifice-type leak passages. Square-edged orifice leaks may not exhibit a maximum flow ratio in the sonic velocity range, as reported by Cunningham[35] in conjunction with his investigation of the discharge characteristics of orifices with supercritical compressible flow, but the effect upon the extrapolation factor is not considered to be significant.

This extrapolation procedure, based on laminar flow, was followed in establishing the allowable leakage rate in the tests conducted for the NASA Plum Brook reactor containment vessel.[36] The maximum allowable leakage rate for this vessel was specified as 0.022% of the initial total weight of contained air per day at the 0.3-psig pressure expected in the event of the maximum credible accident. Since the leakage rate tests were conducted at a 4-psig overpressure, the permissible leakage rate was established as approximately 0.27% per day of the vessel's contained air during test. By using Eq. (10.39), the fact that laminar flow was employed in the extrapolation was verified.

The extrapolation of the measured leakage rate, as determined with air in the containment vessel, represents the value that may be expected at the extrapolated pressure with air at the same test temperature. A correction of the extrapolated leakage rate may then be applied for the accident condition by using the applicable relationships of Eqs. (10.45) through (10.51) if the predominant flow mode of all leaks is known. Unfortunately, the measured leakage rate is not usually the result of one type of flow through all existing leaks, but, rather, a combination of several types of flow. Consequently, any corrections applied by the extrapolation factors of Eqs. (10.45) through (10.51) must be weighed with conservative judgment.

Because of the limited test data available, it is not yet possible to generalize the likely leakage behavior of all containment vessels. It is necessary to conduct leakage-rate tests at several pressure levels

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*$C_{de}/C_{dt}$ has been assumed to be equivalent to 1.

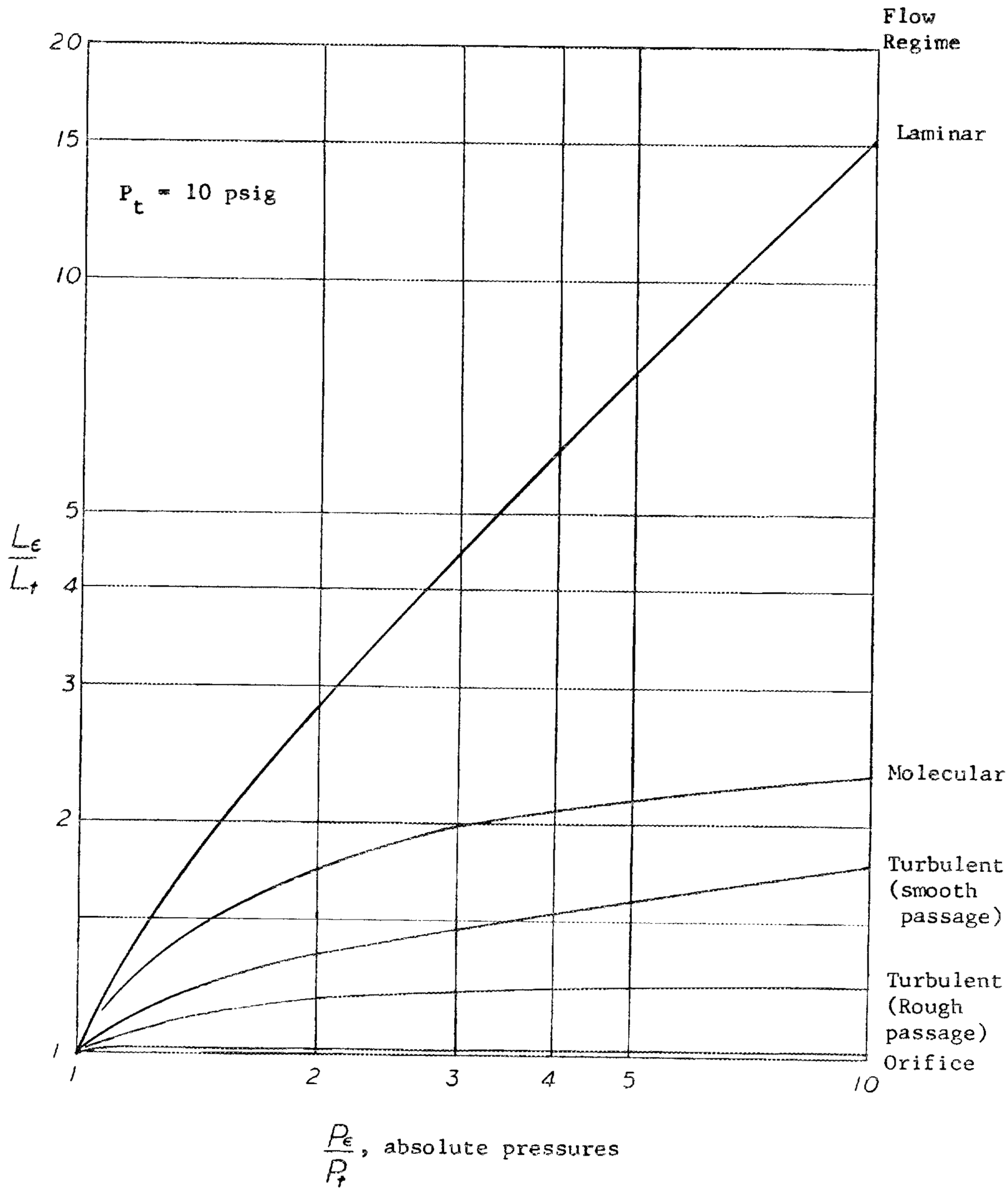


Fig. 10.8. Extrapolation Factor Versus Flow Regime.

to provide a reasonable estimate of the predominant flow mode that characterizes the leakage behavior of the individual containment system. Such test results will, however, be indicative of leakage characteristics at the time of tests. There will still remain some doubt as to whether subsequent tests will demonstrate the same or similar leakage behavior.

### 10.4.6.3 Leakage-Rate Test Data Interpretation

Data from recent containment leakage-rate tests conducted at multipressure conditions were examined to determine what correlations, if any, could be found with respect to predictions made from theoretical considerations. Of the reactor plant containment vessels that have been tested to determine the leakage rate, only relatively few have been tested at several pressure levels. The tests of interest are exemplified by those conducted on the following containment systems:

1. HWCTR[37]
2. NS Savannah[38,39]
3. Agesta Nuclear Power Station[40]
4. EVESR[41]

1. HWCTR Containment Vessel. Leakage-rate tests were performed at 5 and 12 psig on the HWCTR composite steel and concrete containment vessel to verify the assumption that the leakage rate was proportional to the gage pressure, an assumption which was considered to be conservative.[37] The relation between leakage rate and test pressure was found to fall within the bounds of this assumption. Consequently, subsequent retests of the vessel will be performed at 5 psig, and the measured leakage rate will be extrapolated to 24 psig in the direct ratio of the gage pressures.

2. NS Savannah Containment Vessel. Leakage-rate data for the NS Savannah containment vessel, as tested in early 1963, are given in Table 10.4, including the extrapolation factors derived from Eq. (10.39) by assuming that laminar flow predominated for all leaks in this test. From the test results plotted in Fig. 10.9, it may be seen that the measured leakage rate exceeded the leakage rate based on an assumed predominant laminar flow. Apparently, the leakage behavior was influenced by factors

Table 10.4. NS Savannah Leakage-Rate Data

| Test Pressure (psig) | $P_e/P_t$ | Measured Leakage Rate, $L_r$ (% per 24 hr) | Extrapolation Factor Based on Measured Flow, $L_e/L_t$ | Calculated Extrapolation Factor Based On Laminar Flow, $L_e/L_t$ | Modified Factor, $\frac{L_e}{L_t} \frac{P_e}{P_t}$ |
|---|---|---|---|---|---|
| 6 | 1.0 | 0.108 | 1.0 | 1.0 | 1.0 |
| 12 | 1.29 | 0.267 | 2.47 | 1.81 | 2.33 |
| 20 | 1.68 | 0.47 | 4.35 | 2.76 | 4.64 |
| 30 | 2.16 | 0.86 | 7.96 | 3.89 | 8.4 |

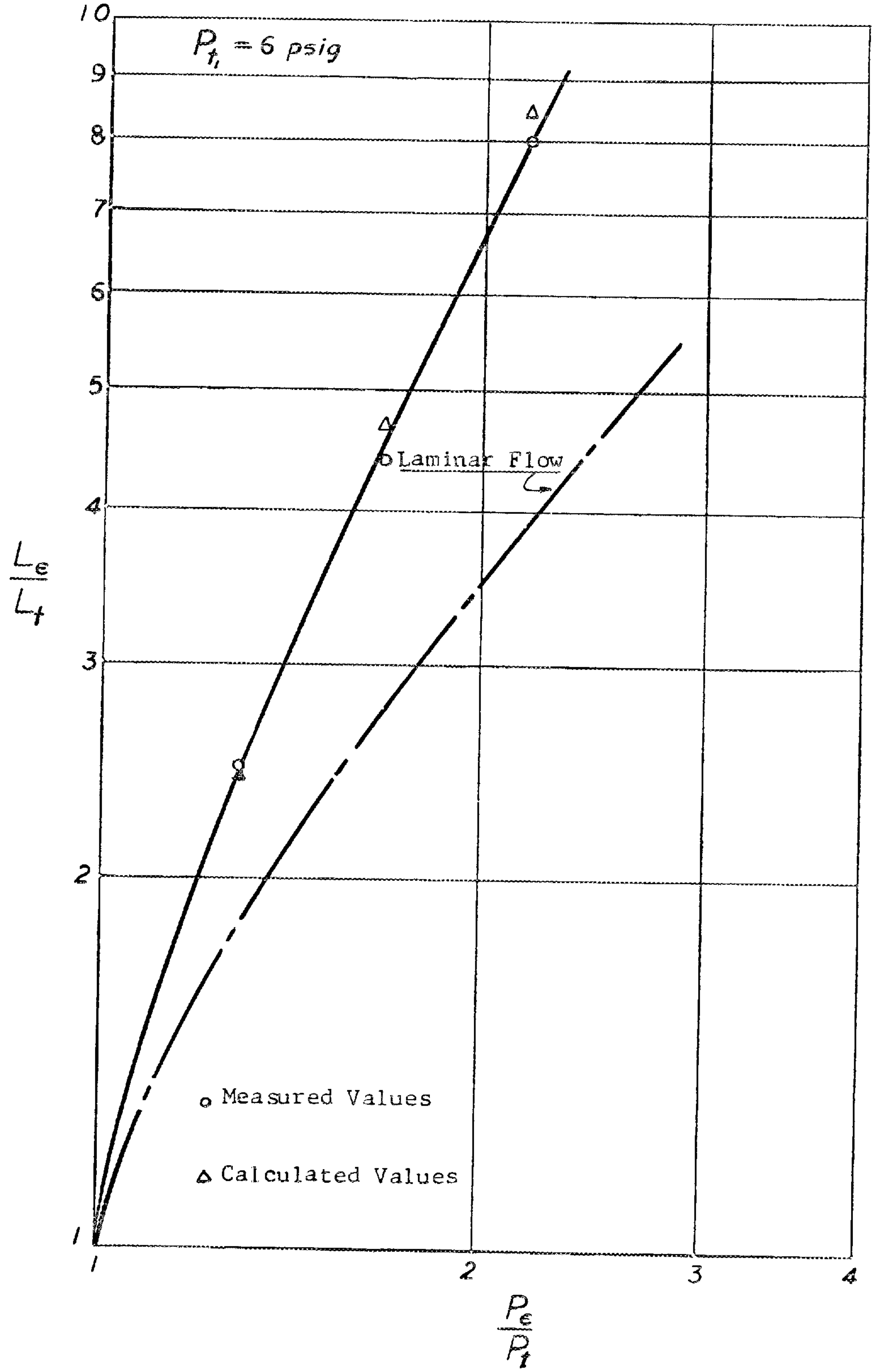


Fig. 10.9. Extrapolation of NS Savannah Containment Vessel Leakage Experience.

other than the pressure parameters that control leakage. If it is assumed that the geometry factor of the leak area is not constant (as originally assumed in the analysis) but, instead, that the leak area varies directly as the containment pressure, the leakage-rate extrapolation factor is modified as follows:

$$\frac{L_e}{L_t} = \frac{\overline{P}_e - \dfrac{1}{\overline{P}_e}}{\overline{P}_t - \dfrac{1}{\overline{P}_t}} \frac{A_e}{A_t} ;$$

but, assuming $A_e/A_t \propto \overline{P}_e/\overline{P}_t$, then

$$\frac{L_e}{L_t} = \frac{\overline{P}_e - \dfrac{1}{\overline{P}_e}}{\overline{P}_t - \dfrac{1}{\overline{P}_t}} \frac{\overline{P}_e}{\overline{P}_t} ,$$

or

$$\frac{L_e}{L_t} = \frac{\overline{P}_e^2 - 1}{\overline{P}_t^2 - 1} . \tag{10.52}$$

The values of this modified extrapolation factor have been computed, as shown in the last column of Table 10.4 and added to the plot of Fig. 10.9. It is interesting to note that this modified extrapolation factor is in close agreement with the actual measured results, and it may therefore be considered as a reasonable upper bound factor for this test. One plausible explanation of this behavior is the effect of pressure on leak geometry, which tends to separate or enlarge a leak gap as the pressure rises. The averaged effects of (1) the predominant leakage behavior through the various leaks and (2) the change in geometry of the leaks with increasing pressure combined to yield the observed leakage-rate results.

Later experience in leak testing of the NS Savannah has shown the extrapolation formula to be that for laminar flow. Equation (10.39) is realistic to about 30 psig; above 30 psig the extrapolation formula for turbulent flow, Eq. (10.40), applies.[39] It was postulated that up to approximately 30 psig the dominant flow was viscous, since major portions of the leakage were detected through electrical penetrations where long, thin, capillary-type leak geometries could exist. It is suspected that above 30 psig the capillary-type leaks reached a critical pressure condition.

3. Agesta Nuclear Power Station Containment Structure. Similar data reported for the Agesta containment structure test are summarized in Table 10.5. This containment vessel consists of a steel lining of an excavated granite-rock underground site. The steel lining provides the containment barrier. Containment leakage-rate tests consisted of measurements at four pressures comparable with the pressures utilized in the NS Savannah tests.

Table 10.5. Agesta Leakage-Rate Data

| Test Pressure (psig) | $P_e/P_t$ | Measured Leakage Rate, $L_r$ (% per 24 hr) | Extrapolation Factor Based on Measured Flow, $L_e/L_t$ | Calculated Extrapolation Factor[a] Based on Laminar Flow, $L_e/L_t$ |
|---|---|---|---|---|
| 4.4 | 1.0 | 0.0815 | 1.0 | 1.0 |
| 14.7 | 1.54 | 0.202 | 2.48 | 2.83 |
| 23.9 | 1.94 | 0.286 | 3.51 | 4.02 |
| 29.8 | 2.33 | 0.363 | 4.46 | 5.05 |

[a]Calculated from Eq. (10.39).

By plotting the values of Table 10.5 as in Fig. 10.10, it becomes apparent that extrapolation factors derived from the measured leakage rates are consistently less than those calculated on the assumption of laminar flow. However, if, in this case, the assumption is made that laminar flow predominated above $P_e/P_t$ ratios of 1.54 (corresponding to the second test pressure), the leakage behavior is definitely within the laminar flow regime. It may be deduced that other than laminar flow existed at test pressures below 14.7 psig but that laminar flow predominated above 14.7 psig. On the basis of this assumption, Table 10.6 demonstrates the close agreement between test and calculated values for test pressures above 14.7 psig.

From the above analysis, it may be concluded that for this containment structure the extrapolation factors based on the laminar flow relationship of Eq. (10.39) provide a reasonable estimate of leakage as a function of pressure. In this instance, it is conceivable that leak geometries varied as the test pressure increased to approximately 14.7 psig, after which, leakage exhibited laminar flow behavior under conditions of constant leak area.

4. ESADA Vallecitos Experimental Superheat Reactor Containment Vessel. A more recent series of tests conducted at two pressure levels, 9.75 and 49.5 psig, on the conventional steel containment building of the EVESR facility[41] yielded the data of Table 10.7. It is especially noteworthy that these leakage-rate tests were preceded by meticulous pretesting of containment components, that is, penetrations, access openings,

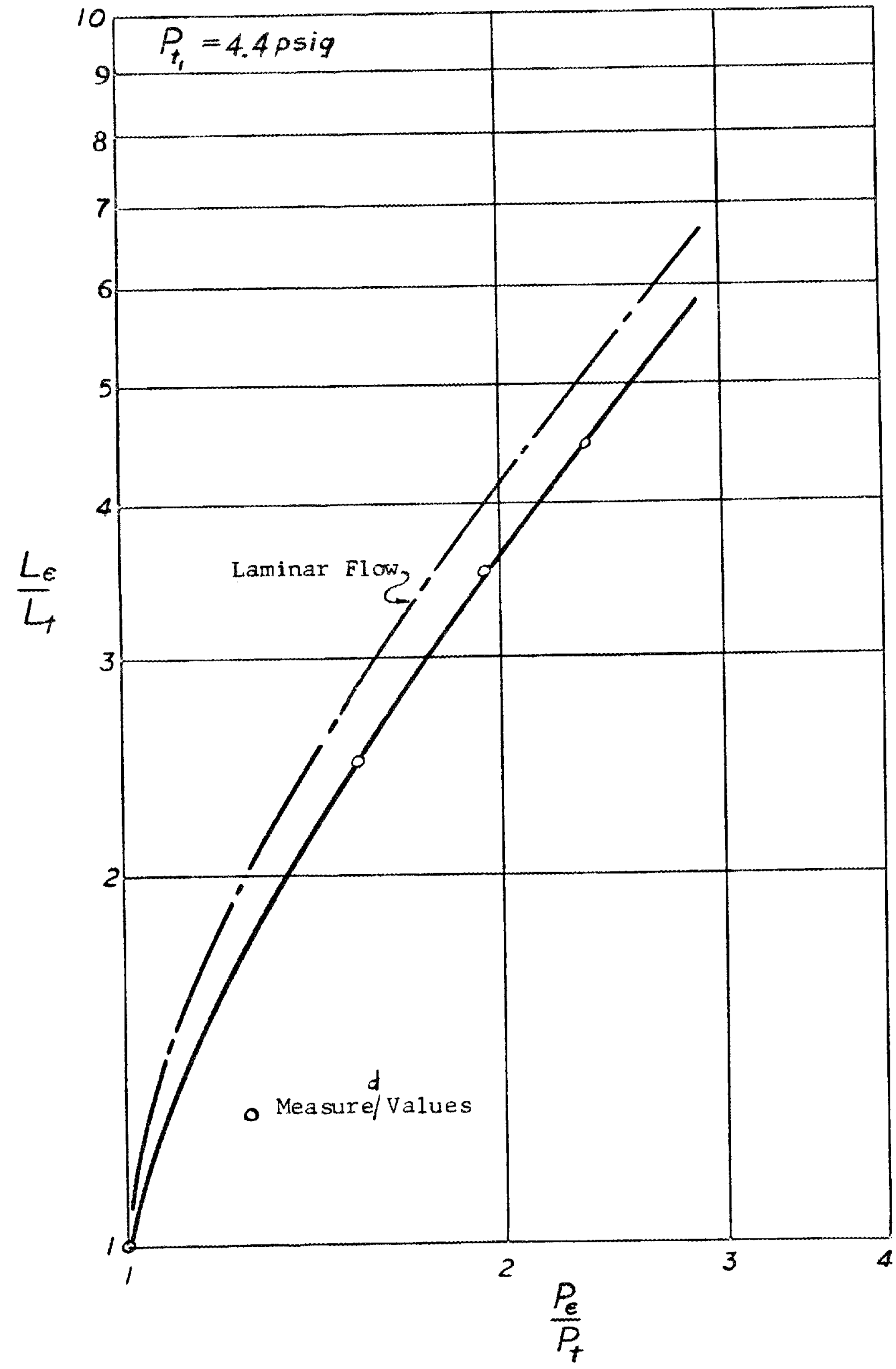


Fig. 10.10. Extrapolation of Agesta Nuclear Power Station Containment Vessel Leakage Experience.

Table 10.6. Agesta Modified Leakage-Rate Data

| Test Pressure (psig) | Measured Leakage Rate, $L_r$ (% per 24 hr) | Extrapolation Factor Based on Measured Flow, $L_e/L_t$ | Calculated Extrapolation Factor[a] Based on Laminar Flow, $L_e/L_t$ |
|---|---|---|---|
| 14.7 | 0.202 | 1.00 | 1.00 |
| 23.9 | 0.286 | 1.415 | 1.42 |
| 29.8 | 0.363 | 1.8 | 1.8 |

[a]Calculated from Eq. (10.39).

Table 10.7. EVESR Leakage-Rate Data

| Test Pressure (psig) | $P_e/P_t$ | Leakage Rate, $L_r$ (% per 24 hr) | Extrapolation Factor Based on Measured Flow, $L_e/L_t$ | Calculated Extrapolation Factor[a] Based on Turbulent Flow (Smooth), $L_e/L_t$ |
|---|---|---|---|---|
| 9.75 | 1.0 | 0.0813 | 1.0 | 1.0 |
| 49.5 | 2.62 | 0.121 | 1.49 | 1.43 |

[a]Calculated from Eq. (10.41).

air locks, isolation valves, and pneumatic instrument lines. All leaks detected during the pretesting phase were repaired and retested prior to the containment building tests. Detailed precautions were taken to precheck all possible leak sources, using either soap bubble or halide leak detection techniques.

The observed low leakage of the containment building is, therefore, not surprising and might even be considered as indicative of the attainable lower bound of leakage for a carefully constructed containment vessel. Insufficient data were collected in performing these tests to ascertain the degree of uncertainty or inaccuracy inherent in the reported leakage-rate values. Notwithstanding, the results do indicate that with proper inspection, careful pretesting of all penetrations and other containment components, and repair of all detected leaks a containment leakage rate of the order of 0.1% per day was achieved. This leakage rate should not be interpreted, however, as the actual leakage rate that existed prior to the pretests and leak repairs, nor should it be assumed that such low values will not change throughout plant lifetime. The results are indicative only of the leakage behavior for the conditions existing at the time of the tests.

The tabulated data are plotted in Fig. 10.11 and compared with the leakage behavior defined by turbulent flow through leaks having smooth passages. Although the results of the composite leakages show a tendency toward turbulent flow characteristics, the performance of only two tests of this containment structure is considered insufficient to interpret the predominant leakage behavior. The results are, therefore, considered of limited value for predicting containment leakage trends.

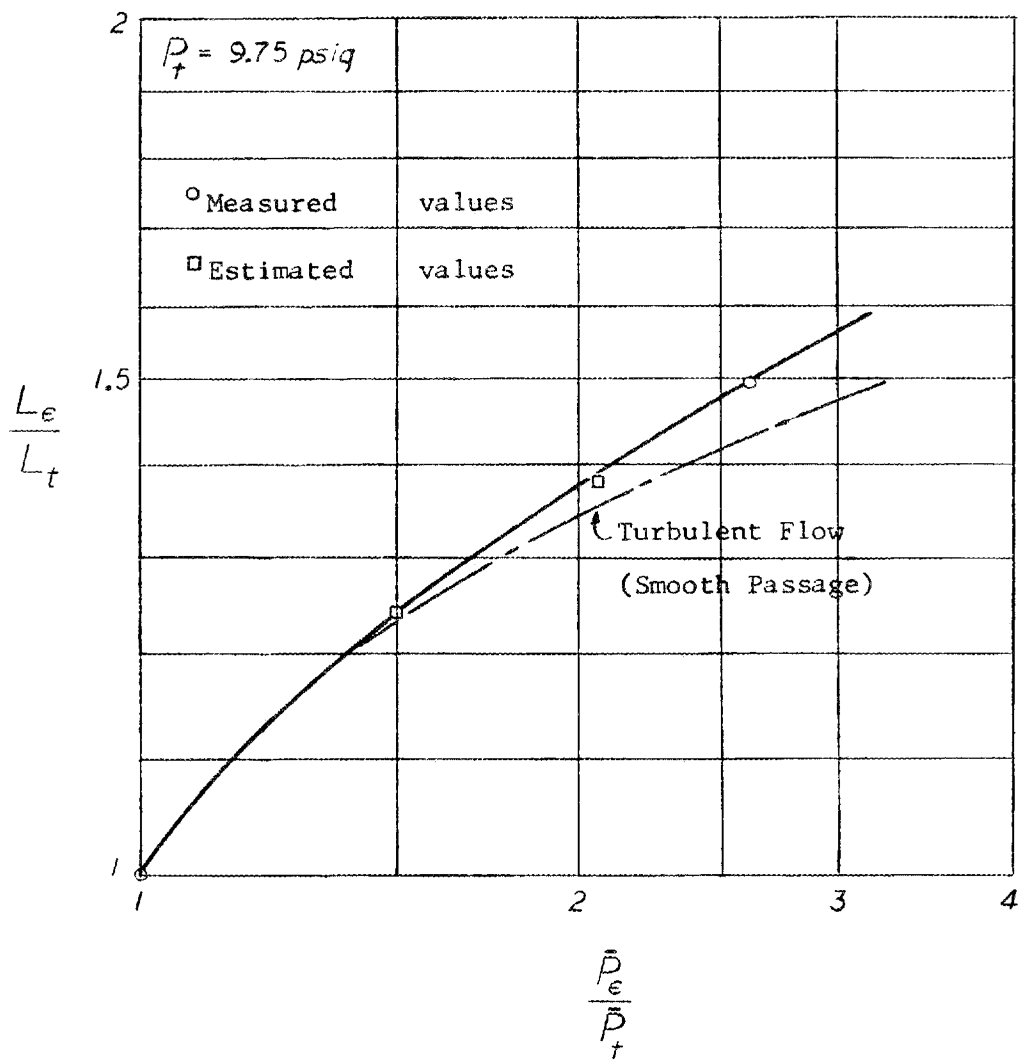


Fig. 10.11. Extrapolation of EVESR Containment Vessel Leakage Experience.

10.4.6.4 Significance of Tests for Leakage Prediction

Reliable prediction of leakage rates at higher pressures from periodic low-pressure tests cannot yet be accepted without reservations in view of the limited test data available on leakage changes during service lifetime of the containment structure. Hence improved leak testing and retesting procedures, such as continuous monitoring systems, will not only obtain more data per se but also enhance the ability to predict vessel leakage characteristics. However, the following conclusions are suggested by the analyses of the containment tests of this section:

1. Until additional testing experience becomes available, conservatism in leakage-rate predictions indicates that extrapolation factors* based on laminar flow should be used and further modified to reflect the changes in leak geometry with pressure increase.

2. Periodic leakage-rate tests should preferably be conducted on the "as is" condition of the containment structure, without any prior leak detection or leak repairs performed directly before the scheduled containment tests. The results thus obtained will represent, in a meaningful manner, the prevailing leakage rate of the containment structure during recent operation of the nuclear power facility; and if early correlation of individual and total leakages is accomplished, the results may indicate what the containment leakage will be in the future.

3. Multipressure leakage-rate tests should be considered as the most practical means, for the present, to establish the leakage characteristic of a containment structure when extrapolation of reduced-pressure leakage data is comtemplated for future tests. Comparing the results of two multipressure leakage-rate tests performed over a reasonable interval of time may provide a measure of the degree of deterioration or significant change in leakage characteristics of the containment structure. Multipressure tests may therefore provide a reasonable basis for periodic reduced-pressure leakage tests, as well as establishing the frequency requirements for these tests during the service lifetime.

4. Any containment design that would permit either individual and local testing of the penetrations, access openings, air locks, etc., at design pressure or incorporate means to continuously monitor the leakage at these same areas without the need for overall containment leakage tests might provide an economical solution to the problem of reduction of periodic testing requirements. With the knowledge of the leakage condition at any time at the most vulnerable areas of the containment structure where leakages develop, and a continuing program of maintenance and leak repair, the frequency of periodic overall containment tests could justifiably be reduced.

The results of the containment building tests were not surprising, and it may be considered that the attainable lower bound of leakage for a properly constructed containment structure has been approached. Insufficient data were collected in performing these tests to ascertain the degree of uncertainty or inaccuracies inherent in the reported leakage-rate values. The results do indicate, however, that with proper maintenance, inspection, and pretesting of containment building components, leakage rates of the order of 0.1% per day can be achieved. These

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*Such factors are intended to apply to upward extrapolation.

values are not to be interpreted as the actual leakage rates that existed prior to the pretests and repairs of the containment components. Leakage rates at least one order of magnitude larger than the reported values are conceivable under conditions where maintenance and inspection are neglected between test intervals. The tests should only be considered as indicative of the containment leakage behavior for the conditions existing at the time of the test. Reliable prediction of leakage rates (at, say, 50 psig) from future periodic low-pressure tests (at, say, 10 psig) cannot yet be justified in view of the limited knowledge of containment leakage changes during service. Until more reliable testing experience becomes available, conservatism in leakage-rate determinations dictates the use of extrapolation factors based on laminar flow.

### 10.4.7 Theory and General Procedure for Pressure-Decay Leakage-Rate Testing

#### 10.4.7.1 Theory

At the ranges of temperature and pressure encountered in containment vessels, air behaves very much like a "perfect gas" and its state in the vessel can be represented by the well-known perfect gas law:

$$PV = wRT \; , \tag{10.53}$$

where

$P$ = absolute pressure, psfa,
$V$ = internal volume of the containment vessel, $ft^3$,
$w$ = weight of air in the vessel, lb,
$T$ = absolute temperature of the air, °R,
$R$ = gas constant for air, ft/°R.

Since $R$ and the containment volume, $V$, are essentially constant, leakage (i.e., a decrease in $w$) can be determined by measuring $T$ and $P$. If $T$ is also relatively constant, leakage is directly proportional to a decrease in pressure. This is the principle of the common "pressure-drop test" used to indicate leakage in many gastight systems and vessels. The pressure-drop test is simple to perform and, since it is an integrated test of the entire system, it is a good qualitative indication of the degree of system leaktightness.

In leakage-rate measurements of reactor containment vessels, however, several difficulties arise that preclude the use of a simple pressure-drop test. These difficulties are due principally to the very low leakage rates usually specified for containment vessels. A low leakage rate requires that the leakage rate test be conducted over a considerable period of time so that the total leakage can be distinguished from normal instrument and measurement inaccuracies. During a long testing period, temperature fluctuations can be considerable and must be taken into account when calculating the leakage rate. This requires that temperature, as well as pressure, be measured accurately during the test. Furthermore, since temperature can vary considerably within the containment vessel itself, temperature measurements usually must be made at several representative locations to obtain a true average air temperature. Thus, measurement of the temperature may be a more difficult problem than measurement of the pressure, which is essentially constant throughout the containment vessel at any given time. Other factors that may affect the leakage-rate test results include a change in humidity of the contained air during the test and expansion of the vessel. These factors are discussed in Section 10.4.12.

#### 10.4.7.2 General Procedure

The initial leakage-rate test of a containment vessel is normally performed after initial local leak detection has been completed, after all detectable leaks have been repaired, after the pressure test has been successfully completed, and while the vessel is still at design pressure. A final leakage-rate test is often performed later after all equipment has been installed in the vessel and before the plant is operated. Both tests follow essentially the same procedure, except that it may not be possible to perform the final test at full design pressure. The test period begins after conditions in the containment vessel have stabilized and usually during the early morning hours when atmospheric conditions are most stable. The test period ends one or more days later at approximately the same time of day so that final conditions are similar to the initial conditions. The initial and final states of the air in the vessel are all that are required to calculate the total amount of leakage during the test period. However, it is customary to record and plot the data periodically during the test so that a trend is established and any spurious data become apparent.

Sections 10.4.8 and 10.4.9 discuss the two principal methods used for containment vessel leakage-rate testing. A step-by-step procedure of a typical initial leakage-rate test is presented in Section 10.5 as part of a typical containment vessel proof-testing sequence.

#### 10.4.7.3 Equipment Required

Some of the equipment required to conduct an integrated leakage-rate test is the same as that required for the containment pressure test, in particular, a compressor or compressors capable of pressurizing the vessel to the design pressure within a reasonable period of time. In addition, blowers for circulating the contained air during the test to prevent thermal stratification are often required. In some cases, the blowers provided for the containment ventilation system may be capable of performing this function. However, it must be kept in mind that blowers not designed to operate at the relatively high test pressure and increased air density can be overloaded by such use, and precautions must be taken to prevent this. Much more sensitive pressure instrumentation is required for a leakage-rate test than is normally used for the pressure test. Accurate temperature instrumentation is also required. A discussion of the instrumentation sensitivity needed is given in Section 10.4.12.5.

### 10.4.7.4 Precautions

The hazards associated with a gas under the pressure that exists for the pressure test also exist for the leakage-rate test, and similar precautions must be followed. Except for low-pressure tests, access to the vessel is not permitted during the test, and no unauthorized personnel are permitted within a specified distance of the vessel. Thus temperature and pressure instrumentation leads must be brought out to a remote location. All equipment that must operate within the containment vessel during the period of the test, such as the blowers that maintain circulation of the air, must be capable of operating at this pressure for long periods of time because of their inaccessibility during the test. Any other equipment or instrumentation within the vessel that could be damaged by the test pressure must be removed or protected before the test is started.

Special precautions must be taken to avoid introducing errors into the leakage-rate measurement due to changes in humidity of the containment vessel atmosphere. If possible, the containment vessel should be pressurized with air of relatively low humidity to avoid moisture condensation within the vessel during the test. The water-vapor content of the containment air should be monitored during the test, and, if a change occurs, the measured leakage rate should be corrected by the methods outlined in Section 10.4.12.1. Water vapor pressure may be determined by direct measurement of the dew point, by measurement of absolute humidity or relative humidity, or by measurement of wet-bulb and dry-bulb temperatures.

## 10.4.8 Absolute Method

### 10.4.8.1 Method

The absolute method of leakage-rate determination is a direct application of the perfect gas law. It requires accurate measurement of the temperature and pressure of the air in the containment vessel at the beginning and at the end of the test period. The total leakage of air during the test period is then the difference in weight of air in the vessel, w, between the beginning and the end of the test. Denoting conditions at the beginning of the test with a subscript 1 and conditions at the end of the test with subscript 2,

$$P_1 V = w_1 R T_1 \ , \tag{10.54}$$

and

$$P_2 V = w_2 R T_2 \ . \tag{10.55}$$

The total leakage during the test period is then

$$w_1 - w_2 = \frac{V}{R}\left(\frac{P_1}{T_1} - \frac{P_2}{T_2}\right) , \tag{10.56}$$

or

$$w_1 - w_2 = \frac{P_1 V}{RT_1}\left(1 - \frac{P_2 T_1}{P_1 T_2}\right) , \tag{10.57}$$

or

$$w_1 - w_2 = w_1 \left(1 - \frac{P_2 T_1}{P_1 T_2}\right) . \tag{10.58}$$

The fractional leakage is

$$\frac{w_1 - w_2}{w_1} = 1 - \frac{P_2 T_1}{P_1 T_2} . \tag{10.59}$$

The percent leakage per 24 hr, or leakage rate, designated $L_r$, is expressed as follows, where H is the number of hours of the test period:

$$L_r = \frac{24}{H}\left(1 - \frac{P_2 T_1}{P_1 T_2}\right) 100 . \tag{10.60}$$

In the event that the initial and final temperatures are the same, the leakage-rate expression can be simplified, as follows:

$$L_r = \frac{24}{H}\left(1 - \frac{P_2}{P_1}\right) 100 . \tag{10.61}$$

The principal difficulty and major source of error in determining the leakage rate is that of obtaining an accurate, truly average temperature for the total volume of air in the containment vessel. Not only will the average temperature vary throughout the test period but the spatial temperature distribution within the containment vessel at any one time will vary as well. For example, as the sun comes up in the morning, the eastern side of the containment vessel will be heated more than the western side, and thus there will be a different temperature distribution than will exist later in the day or during the night. Therefore, it is important that sufficient temperature measurements be taken to adequately represent the entire volume of air. If pockets or "cells" of air exist in the containment vessel, each of these should contain a temperature-measuring device, and the temperature reading from each cell should be weighted by the approximate volume of the cell so that a true weighted average temperature is obtained. The temperature variations throughout the vessel can be reduced by circulating the containment air during the test using the normal containment ventilation system blowers or temporarily installed blowers. Circulating the air also improves heat transfer to the temperature-measuring instruments and makes humidity measurements more reliable. In order to measure leakage rates as low as those often specified (0.1% per day), temperature-measuring instruments should be reproducibly readable to approximately 0.2°F. The temperature instruments should be calibrated over the range of temperatures anticipated during the test.

Pressure measurements usually need to be made at only one location in the containment vessel, since static pressure variations within the vessel will be slight and can be neglected in most cases. If circulation of the air in a vessel containing several compartments can produce pressure differentials, however, it may be desirable to have pressure taps in each compartment. Pressure-measuring devices should be reproducibly readable in the range of 0.2 to 0.5 mm Hg (depending on the test pressure) in order to measure the low leakage rates often specified.

### 10.4.8.2 Experience

The absolute method was used in the leakage-rate test of the containment sphere at West Milton, New York.[42] Prior to the test, a preliminary temperature survey was conducted to indicate the feasibility of using this method and to determine the proper location of the temperature sensors so that a representative temperature within the sphere could be obtained. Twenty thermistors were positioned throughout the sphere, and temperature data from these thermistors were recorded hourly over a 14-hr period. This survey showed that temperature equilibrium within 1°F was established throughout the sphere and maintained between midnight and 8:00 a.m. Therefore, for the leakage-rate test itself, only the readings from the thermistor at the geometric center of the sphere were used in the leakage-rate calculations. In this test, the ratio of pressure to temperature, which is proportional to the weight of the gas within the containment vessel, was calculated for each hour of the 30-hr test period and plotted as a function of time. The resulting curve varied within a band of 0.2%, and it was concluded that "the leakage, if any, was not in excess of 0.2 percent in 48 hours of the contents of the sphere pressurized to 19 3/4 psig." Since the specified leakage rate for this containment vessel was 1% in 48 hr, the results were considered to be satisfactory.

The absolute method was also used in two separate leakage rate tests of the EBWR containment vessel.[43] In the first test, ten thermocouples were located on the inside surface of the shell, eight resistance thermometers were suspended in the air space throughout the vessel, three dew-point cells were placed in the air space to determine the approximate relative humidity of the contained air, and six blowers were placed at various locations inside the shell to circulate the air. The leakage-rate test was run at approximately 15 psig. Temperature and pressure readings were taken over a period of eight days at approximately 2-hr intervals. From the temperature and pressure readings, the volume of dry air at standard temperature and pressure within the building was calculated, taking into account variations in water-vapor pressure and variations in the containment vessel volume due to thermal expansion and contraction. Because several erratic readings occurred when the shell temperature was high or was changing rapidly, no readings were considered valid when the shell metal temperature was higher than 50°F or when the average shell metal temperature differed by more than 4°F from the temperature read 2 hr before or after the reading in question. Although there was considerable scatter in the data, the leakage rate was well below the specified allowable leakage rate of 500 scf in 24 hr. After all construction work had been completed, the final leakage-rate test was conducted using somewhat more sensitive instrumentation. The results of this final test showed much less scatter and indicated a leakage rate of approximately 450 scf for 24 hr as compared with the allowable leakage rate for the completed vessel with all equipment installed of 1000 scf per 24 hr (0.25% of net building volume).

The absolute method was also used for tests of the vessels of two Danish reactors,[44] DR 2 and DR 3. The DR 2 is a water-cooled and -moderated 5-Mw pool-type research reactor. Its containment vessel is a steel cylinder 80 ft in diameter and approximately 80 ft high designed for a maximum internal pressure of 2 psig. The DR 3 is a heavy-water-moderated and -cooled 10-Mw research reactor contained in a cylinder 70 ft in diameter and approximately 75 ft high designed for a maximum internal pressure of 6.5 psig. Preliminary and final leakage-rate tests were run on both containment vessels. At DR 2, the average pressures at which the leakage-rate tests were conducted were 1.4 psig and 1.1 psig for the preliminary and final tests, respectively, while at DR 3 the pressure was 5 psig for both tests. To calculate the leakage rate in all these tests, the ratio of the dry air pressure (the total containment vessel pressure minus the water-vapor pressure) to the average containment air temperature was plotted as a function of time. Total pressure was measured by means of a precision barometer and a water manometer, both corrected for thermal expansion. Water-vapor pressure was determined using dry- and wet-bulb thermometers at DR 2 and using humidity elements measuring either the relative humidity or the absolute humidity at DR 3. Several thermistors located throughout the vessels were used to determine average temperatures. At DR 3, 20 thermistors were used. The resulting leakage rates, scaled up to the maximum design pressure of the vessels in direct proportion to the pressure, were estimated to be 0.2% per 24 hr for DR 2 and 0.02% per 24 hr for DR 3. The uncertainty in the DR 3 test results was estimated to be 0.01% per 24 hr. Petersen and Vinther[44] state that they believe the absolute method to be the most convenient of the methods considered. During the initial test at DR 2, two other methods, one using a flask of dry air as a reference vessel and the other measuring air density by weighing a glass ball, were tried; however, neither of these methods was more accurate than the direct method and both were more laborious according to the authors.

In the leakage-rate tests of the Plum Brook Reactor containment vessel[9] the absolute method was used in parallel with the reference vessel method to provide a comparison of the two methods (see Sec. 10.4.11). The allowable flow rate specified for the containment vessel, which has a volume of 500,000 ft$^3$, is 115 scf per day at a pressure of 0.3 psig. However, to accelerate the leakage-rate tests, they were conducted at 4 psig, at which pressure the allowable flow rate is 1530 scf per day. Corrections were made for changes in water-vapor pressure of the containment air and for changes in the containment vessel volume due to changes in the level of the shielding-pool water. Each of the three tests was 50 to 60 hr in duration. The leakage rates measured with the absolute method were 0.088 ± 0.014, 0.170 ± 0.027, and 0.173 ± 0.032% per day for the first, second, and third tests, respectively.

In the leakage-rate tests of the containment vessels for the Dido and Merlin reactors at Julich, Germany,[45] the absolute method was used after studies indicated this method to be less complex for the conditions of the tests. The leakage rate of the Dido containment vessel was determined to be 0.13 ± 0.052% per day during a 48-hr test. In a 72-hr test the leakage rate of the Merlin vessel was found to be 0.043 ± 0.035% per day.

## 10.4.9 Reference Vessel Method

### 10.4.9.1 Method

The reference vessel method for leakage-rate testing of containment vessels was devised to eliminate the necessity for precise measurement of absolute pressure and for multiple measurement of temperature. This method depends on the change in pressure of a constant volume of containment air compared with that of a closed reference vessel, or system of vessels, located within the containment vessel. Leakage of air out of the containment vessel is indicated by a change in differential pressure between the reference system and the containment vessel. This differential pressure is small and can be measured with a water manometer so that greater precision in pressure measurement can be obtained for a given error in linear measurement than is obtained with the mercury manometers used to measure absolute pressure. Because the reference system is located within the containment vessel, the temperature of the containment air and the reference system air should be the same, and temperature compensation should, therefore, be automatic provided humidity corrections are made within the reference system.

There are some complications in using the reference vessel method, however, so not all the apparent advantages of the method can be realized. For example, the problem of properly locating the reference vessels within the containment vessel so that the containment atmosphere is properly sampled and true temperature compensation is achieved is similar to the problem of locating the temperature sensors in the absolute method. This is particularly difficult if the containment vessel is highly compartmentalized, since a separate reference vessel of the proper size and shape is required for each compartment. In addition, when there are relatively rapid temperature changes during the test period, the temperature of the reference system may lag significantly behind the temperature of the containment vessel air if the heat transfer between the containment vessel and the reference system is poor. A test on the NS Savannah reference system, which contains five 8-in. cylinders with 1/4-in. walls, indicated a heat transfer time constant of greater than 2 hr. An additional complication for the reference vessel method over the absolute method is the requirement that the reference system be extremely leaktight. Thus it must be carefully fabricated and tested prior to the leakage-rate test.

A reference system with sufficient leaktightness has not been too difficult to achieve, however, in most tests where this method has been used.

Some of the difficulties of the reference vessel method can be at least partially overcome by plotting the temperature and differential pressure data obtained as a function of time. In this way, any leakage in the reference system or a lack of temperature compensation will become readily apparent. Also, thermal lag will be apparent, and, in some cases, it may be possible to apply suitable corrections to compensate for it. Although thermal lag may cause the differential pressure to vary over a wide range throughout the day, the variation should be similar from day to day and thus can be accounted for on the basis of diurnal temperature changes. It is customary, as it is with the absolute method, to begin and end the test in the early morning hours to take advantage of the relatively stable atmospheric conditions at that time of day.

An analysis of the problem of thermal lag in a reference vessel was made in connection with the leakage-rate tests on the Plum Brook reactor containment vessel.[36] This analysis was performed to determine the magnitude of the temperature difference between the air in the reference system and in the containment vessel under typical test conditions for various reference vessel diameters. For an assumed sinusoidally varying temperature with an amplitude of 8°F and a period of 8 hr, the temperature difference due to phase shifting between the surface and centerline temperatures of the reference vessel was calculated to be 2.2°F for a 24-in.-diam vessel, 0.7°F for a 12-in.-diam vessel, and less than 0.01°F for a 2-in.-diam vessel such as that used for the tests. These temperature differences would result in possible errors in the calculated leakage rate of 280, 89, and 1.4% for the 24-, 12-, and 2-in.-diam vessels, respectively, for the assumed test conditions. A smaller diameter reference vessel would reduce the thermal lag even further, but there would still be errors due to incomplete sampling of the containment atmosphere. Also, changes in the reference system volume due to changes in the manometer fluid level could become more significant as the diameter is decreased (see Sec. 10.4.12.2.)

Starting with the perfect gas law and denoting reference vessel conditions with primes, the leakage rate for the reference vessel method is derived as follows:

1. In the reference vessel,

$$P_1'V' = w'RT_1' \tag{10.62}$$

and

$$P_2'V' = w'RT_2' \ , \tag{10.63}$$

where $w' = w_1' = w_2'$.

2. In the containment vessel,

$$P_1 V = w_1 R T_1 \tag{10.64}$$

and

$$P_2 V = w_2 R T_2 \ . \tag{10.65}$$

The pressure differential between the reference system and the containment vessel at the beginning of the test is then

$$\Delta P_1 = P_1' - P_1 \ . \tag{10.66}$$

Substituting the values of $P_1'$ and $P_1$ from Eqs. (10.62) and (10.64), respectively, into Eq. (10.66) gives

$$\Delta P_1 = R\left(\frac{w'T_1'}{V'} - \frac{w_1 T_1}{V}\right) . \tag{10.67}$$

By transposition,

$$w_1 = \frac{V}{T_1}\left(\frac{w'T_1'}{V'} - \frac{\Delta P_1}{R}\right) . \tag{10.68}$$

Similarly,

$$\Delta P_2 = P_2' - P_2 = R\left(\frac{w'T_2'}{V'} - \frac{w_2 T_2}{V}\right) \tag{10.69}$$

and

$$w_2 = \frac{V}{T_2}\left(\frac{w'T_2'}{V'} - \frac{\Delta P_2}{R}\right) . \tag{10.70}$$

Subtracting Eqs. (10.68) and (10.70) and rearranging terms gives the total leakage during the test period:

$$w_1 - w_2 = \frac{Vw'}{V'}\left(\frac{T_1'}{T_1} - \frac{T_2'}{T_2}\right) + \frac{V}{R}\left(\frac{\Delta P_2}{T_2} - \frac{\Delta P_1}{T_1}\right) . \tag{10.71}$$

Substituting for $w'$ from Eq. (10.62) then gives

$$w_1 - w_2 = \frac{P_1'V}{RT_1'}\left(\frac{T_1'}{T_1} - \frac{T_2'}{T_2}\right) + \frac{V}{R}\left(\frac{\Delta P_2}{T_2} - \frac{\Delta P_1}{T_1}\right). \qquad (10.72)$$

Dividing this by $w_1$ from Eq. (10.64) gives the fractional leakage during the period of the test as

$$\frac{w_1 - w_2}{w_1} = \frac{P_1'T_1}{P_1T_1'}\left(\frac{T_1'}{T_1} - \frac{T_2'}{T_2}\right) + \frac{T_1}{P_1}\left(\frac{\Delta P_2}{T_2} - \frac{\Delta P_1}{T_1}\right). \qquad (10.73)$$

If, as is assumed with the reference vessel method, the temperatures in the reference system and the containment vessel are the same, i.e, that $T_1 = T_1'$ and $T_2 = T_2'$, Eq. (10.73) reduces to

$$\frac{w_1 - w_2}{w_1} = \frac{T_1}{P_1}\left(\frac{\Delta P_2}{T_2} - \frac{\Delta P_1}{T_1}\right). \qquad (10.74)$$

The leakage rate per 24 hr is then expressed as follows, where H is the number of hours of the test period:

$$L_r = \frac{24}{H}\frac{1}{P_1}\left(\frac{\Delta P_2 \; T_1}{T_2} - \Delta P_1\right) 100 . \qquad (10.75)$$

This is the equation contained in the proposed ANS standard for containment vessel leakage rate testing[1] using the reference vessel method.

With the further assumption that the temperature at the beginning of the test and at the end of the test are the same, i.e., that $T_1 = T_2$, the leakage rate equation simplified to

$$L_r = \frac{24}{H}\frac{1}{P_1}(\Delta P_2 - \Delta P_1)\; 100 . \qquad (10.76)$$

The fact that this simplified equation has no temperature terms should not be construed to mean that the temperature measurements need not be taken, for by no other method may it be proved that $T_1$ does equal $T_2$ and that the equation is valid.

In the event that the differential pressure at the beginning of the test is 0 (i.e., that $\Delta P_1 = 0$), the leakage rate is

$$L_r = \frac{24}{H}\frac{\Delta P_2 \; T_1}{P_1T_2}\; 100 , \qquad (10.77)$$

or, if in addition $T_1 = T_2$, the leakage rate is simply

$$L_r = \frac{24}{H} \frac{\Delta P_2}{P_1} 100 \ . \qquad (10.78)$$

Any of the above equations for leakage rate may be used; however, the simplifying assumptions inherent in each form of the equation should be realized. It is to be noted that all the above equations for the leakage rate contain the absolute pressure, $P_1$, in the denominator. This might seem to refute the advantage claimed for the reference vessel method that only differential pressure measurements are required. However, since absolute pressure enters the equation only as a multiplier to yield percent leakage rather than the total amount of leakage, the accuracy of the measurement of absolute pressure is not critical. An error in absolute pressure will result in the same percentage error in the percentage leakage rate, and even the maximum errors occurring in ordinary pressure measurements are not large percentage errors and are not significant unless very high accuracy is required.

10.4.9.2 Experience

The reference vessel method was first used in December 1956 for the leakage-rate test of the containment vessel for the Vallecitos Boiling-Water Reactor (VBWR) in Pleasanton, California.[46] This containment vessel has a design pressure of 45 psig, and the allowable leakage rate was specified as 1% of rated pressure in 24 hr. The reference system consisted of three 55-gallon tanks that were evenly spaced within the containment vessel. The test was conducted at a pressure 26.5 psig, and readings were taken for a period of about 2 1/2 days. During the night hours, when there was little variation of temperature within the vessel, the differential pressure readings were about 1/2 in. $H_2O$ for three consecutive evenings. Since these readings were estimated to be reliable to ±1/8 in. $H_2O$, it was concluded that the leakage did not exceed 0.02% per day. It is pointed out that the pressure in the vessel varied as much as +6% to −1 1/2%. However, the manometer differential pressure varied only about ±1/2%, indicating that there was excellent compensation for pressure changes caused by temperature and that there was no measurable leakage from the vessel.

A subsequent leakage rate test of the VBWR containment vessel was conducted in August 1957, after all equipment had been installed.[46] This test was conducted at a reduced pressure of 12 psig to prevent possible pressure damage to the installed equipment, but the test procedure was otherwise similar to that of the first test. The results of this test again demonstrated that the leakage rate was well below the allowable limit. It was noted that the temperature compensation was even better in the second test than it was for the first. This was attributed to the large mass of concrete in the vessel, which would tend to decrease the inside temperature variations. Because the measured leakage rates were so low and because the condition of the vessel was different in each test, the results of the two tests cannot be compared directly.

Since the time of the VBWR containment vessel leakage-rate test, most of the reactor containment vessels built in this country have been tested using the reference vessel method. The containment vessels for the following plants have been tested in this manner:[47]

1. Air Force Nuclear Engineering Test Reactor at Wright-Patterson Air Force Base, Dayton, Ohio; tested in January 1958.
2. Dresden Nuclear Power Station of the Commonwealth Edison Company, Dresden, Illinois; tested initially[48] in March 1958; periodic retest in October 1961.
3. Enrico Fermi Atomic Power Plant of the Power Reactor Development Corporation, Lagoona Beach, Michigan; tested in December 1958.
4. Indian Point Plant of the Consolidated Edison Company of New York; tested initially in May 1959; final test in May 1962.
5. Elk River Plant of the Rural Cooperative Power Association in Minnesota; tested initially in May 1959; final test in January 1961.
6. Yankee Atomic Electric Company Plant in Rowe, Massachusetts; tested[49] in July 1959.
7. Big Rock Point Plant of Consumers Power Company at Charlevoix, Michigan; tested initially[50] in January 1961; final test in June 1963.
8. BONUS Reactor operated by the Puerto Rico Water Resources Authority; initial test in May 1961; final test in August 1963.

In many of these tests, the leakage rate was indicated to be less than 0.05% per day and as low as 0.01% per day. The initial test was made on an empty containment vessel. Where internal concrete, compartments, piping, etc. are present for a final or periodic test, the measurement is more complicated and higher leakage rates should be expected. Inherent errors and uncertainties in leakage-rate measurement can readily exceed, by an order of magnitude, leakage rates approaching 0.01%.

## 10.4.10 Other Methods

Although the absolute method and the reference vessel method have been used for leakage rate testing of nearly all containment vessels built to date, other methods have been proposed and may be preferable in some circumstances. Brittan[18] has briefly described several of these alternate methods.

### 10.4.10.1 Measurement of Makeup Air

Leakage can be determined by measuring the amount of air that must be added to the containment vessel to maintain the pressure at its initial value. As described by Brittan, this method, proposed at one time for the EBWR, would use makeup air supplied from a compressed air cylinder placed on a platform scale to determine the weight of air released. A rotary-type flow meter was also to be used as a check on the scale readings. This method also requires reliable monitoring of air temperature so that the initial conditions can be duplicated as closely as possible.

1. Yankee Nuclear Power Station. A method similar to the makeup air method is used at the Yankee Nuclear Power Station both as a secondary method to the reference vessel method and for periodic checks of leakage rate during plant operation.[51] For leakage-rate testing with the plant shut down, the containment vessel is pressurized to 15 psig and any leakage is indicated by a change in the differential pressure between a permanently installed reference system and the containment vessel, as is normally done using the reference vessel method described in Section 10.4.9. In addition, a standard gas flowmeter can be used to meter the air required to equalize the two pressures. The volume of air added is then equal to the total leakage during the test period. The same leakage-measuring system can be used to indicate any increases in leakage during operation and to guard against the chance of gross leakage through improper closure of openings. During operation, access to the vessel is not normally permitted, and a nominal internal pressure is maintained. Thus it is possible to monitor leakage during operation by periodically adding air through the gas meter until the differential pressure between the containment vessel and the reference system is back to its initial value. Because of the long period of time over which the leakage can be determined, a good indication of the containment vessel leakage rate at operating pressure can be obtained. Because of the difficulties of scaling the leakage rate with pressure, this method may not give a true measure of the leakage rate at design pressure; however, it provides a good check to assure that all openings have been closed and that containment integrity is being maintained.

2. NS Savannah. A similar system is provided on the NS Savannah to periodically check containment vessel leakage.[52] Containment vessel leakage is particularly critical in the case of the Savannah because this plant can be brought into metropolitan areas and because there is greater possibility that leaks may develop as a result of motion-induced stresses. This continuous leakage monitoring system has not given reliable results in the past, principally because air and nitrogen can leak into the vessel from the pneumatic equipment and the hydraulic control-rod-drive system accumulators located within the containment vessel. These leaks can cause the monitoring system to indicate a leakage rate that is lower than the actual rate; thus, tests using the installed reference system have had to be run periodically when the plant has been shut down.

3. Humboldt Bay. A similar process has also been proposed for use on the Humboldt Bay pressure-suppression containment system as a means of continuously monitoring the leakage rate. However, in this case, separate information is desired concerning the leakage rates in the dry well and in the suppression chamber. Since these chambers are interconnected by the vent piping system, it is impossible to pressurize one chamber more than the other, except for the differential water level leg that may be maintained in the submerged vent piping. Additionally, providing any substantial pressure on either of the chambers would significantly increase the contained air mass and reduce the pressure-suppression effect. It has therefore been proposed that the system operate with the cover air pressurized to 10 to 20 in. $H_2O$, with the pressure maintained at a constant value by the periodic addition of a metered quantity of makeup air. Such a system is quite sensitive to operating condition variables due to the high thermal and humidity gradients that exist in the overall system,

but the bulk of information obtained on a continuous long term basis should provide a good indication of any changes in the leakage rate. With sufficient experience, the system will perhaps be useful in predicting the leakage rates that would occur at the design pressures of the respective chambers provided an adequate extrapolation means is determined.

### 10.4.10.2 Adulterant Gas Method

An adulterant gas may be introduced into the vessel and its escape detected and measured by various means. Adulterant gases have been used in many leak testing applications, particularly for local leak detection and location (see Sec. 10.3.5), but they have not been used for integrated leakage-rate testing of containment vessels. For locating leaks, the outside of the vessel is scanned by a detector that is sensitive to the presence of the adulterant gas. For measuring the total integrated leakage rate, it has been proposed that an envelope be placed around the entire vessel to collect all leakage and then measure the concentration of adulterant gas in the air collected. This method may be appropriate for containment vessels that are built with an outside shield or with a secondary containment structure in which all leakage through the primary containment vessel is collected. For example, for a $5 \times 10^6$-$ft^3$ vessel surrounded by a secondary shell with a 2-ft wide annular space, and assuming an internal helium concentration of $10^{-3}$ parts by volume, a helium mass spectrometer sensitivity of $5 \times 10^{-8}$ cc/sec, and complete mixing with the air in the annular space, an integrated leakage rate of approximately 50 $ft^3$ per day or 0.001% per day should be measurable. It also may be possible to sample the atmosphere inside the containment vessel and measure the reduction in concentration of the adulterant gas. Adulterant gases that could be used for this purpose include helium, freon, and certain radioactive gases.

With the adulterant gas method it may be possible to conduct leakage-rate tests at low pressures. This is particularly significant in the case of retests after all equipment has been installed and where some of this equipment cannot be exposed to high pressures. In this case, the difficulty of scaling a leakage rate determined at low pressure to an equivalent rate at a higher pressure is the same as that encountered in all low-pressure leakage-rate tests.

### 10.4.10.3 Checking Individual Penetrations

If it can be assumed that all leakage comes from identified sources, the total vessel leakage rate can be determined by adding up the leakage from each source.[18] This requires that the vessel first be thoroughly tested and all welded joints determined to be sound so that any subsequent leakage occurs only at penetrations. Any of the methods considered for testing should preferably provide for uninterrupted operation of the generating equipment.

One method is to build a leaktight box at each penetration on the outside of the containment vessel. Then the pressure in the box can be reduced, the box can be sealed, and the pressure buildup in the box due to leakage through the penetration can be determined for a short period of time, with 10 min usually being sufficient. All leakages are added and compared with the total weight of air in the vessel to give the leakage rate. An alternate procedure is to pressurize the leaktight box and determine leakage through the penetration by measuring the rate of decrease of pressure. This would permit testing at design pressure and would avoid the uncertainties of seal leakage increases with pressure. The sensitivity of this test may be one or two orders of magnitude greater than that of the integrated leakage-rate test. The main advantage of this method is that it is not necessary to pressurize the vessel and thus it is possible to determine the leakage rate at any time. It is also possible to monitor the leakage rate continuously. The major disadvantage is the uncertainty that all possible leaks in the containment vessel are being monitored.

Another method of checking individual penetrations is the soap bubble growth-rate method. This is an extension of the common soap bubble technique for locating leaks. It requires that all leaks are known and that all leakage occurs at such a rate that the growth of a soap bubble can be timed. Large leaks would break through the soap bubble too quickly to be measured. Brittan[53] points out that a soap bubble growing to a 2-in. diameter in 6 sec represents a leakage rate of 15 $ft^3$ per day and that 100 such leaks would correspond to a leakage rate of 0.025% per atmosphere overpressure on a $3 \times 10^6$-$ft^3$ vessel. This method may be quantitatively more accurate than an integrated leakage-rate test, but it also suffers from the uncertainty that all leaks are being monitored.

#### 10.4.10.4 Resistance Thermometry

In the most recent leak test of the Plum Brook reactor, a 550-ft length of 0.025-in.-diam bare nickel wire was distributed throughout the containment vessel for use as a resistance thermometer. A test of this type combines many of the best features of the absolute and reference methods. Sampling is greatly improved, and at the same time the complexity of the temperature-measuring system is substantially reduced, since only a single resistance readout is required. Furthermore, preparation and installation were found to take only a fraction of the time required for the reference method. Figure 10.12 presents a comparison of the data from the use of a distributed resistance thermometer with a single-point resistance thermometer and the absolute and reference methods.

### 10.4.11 Comparison of Leakage-Rate Testing Methods

The selection of one or more of the leakage-rate testing methods discussed above involves consideration of many factors, including the applicability of the test method to the containment design at hand, the sensitivity of the test required, the environmental effects on the system being tested, the time and personnel training required, the cost and availability of any specialized equipment required, and the possible

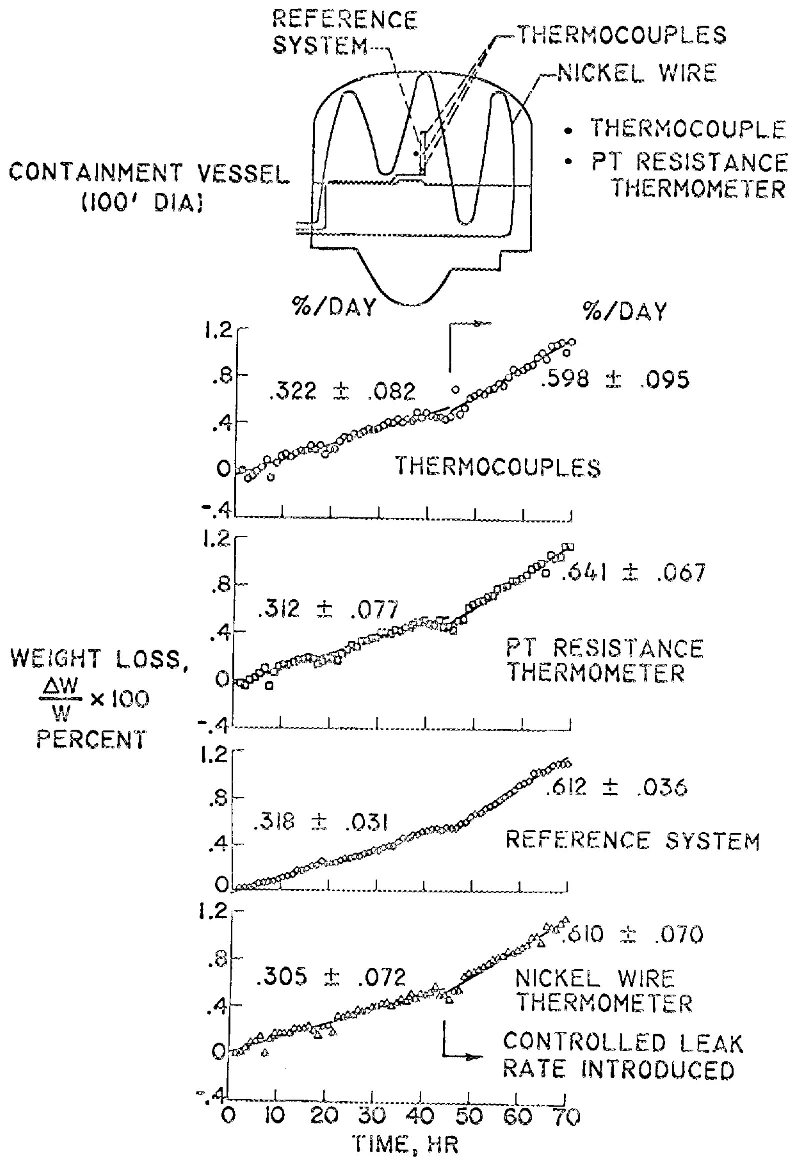


Fig. 10.12. Plum Brook Reactor Leakage Rate Results Obtained with Four Different Methods of Temperature Measurement.

future usability of the equipment and experience in retesting work. In certain special cases one or more of the methods discussed in Section 10.4.10 may be used. For example, the primary barrier of a double-barrier containment system may be tested with helium, as described in 10.4.10.2, if a very low maximum allowable leakage rate is specified. For testing a relatively small and easily accessible vessel with only a few penetrations, or for testing only a portion of a larger system, one of the local leak testing techniques adaptable to integrated leakage-rate testing may be used. For the most part, however, these methods have not been used for leakage-rate testing of vessels as large as most containment vessels, so they would not be appropriate for leakage-rate proof testing without further demonstration of their suitability. In the large majority of cases, therefore, the selection will be between the absolute and the reference vessel methods for the integrated leakage-rate testing of typical low-leakage containment vessels.

The principal U.S. power and test reactor facilities for which integrated leakage-rate tests of the containment vessel have been conducted are listed in Table 10.8. The table indicates the test pressure, the

Table 10.8. Integrated Leakage-Rate Tests of Various U.S. Reactors[a]

| Reactor | Type of Test | Test Pressure (psig) | Observed Leakage Rate (% per day) | |
|---|---|---|---|---|
| | | | Absolute Method | Reference Vessel Method |
| Test reactors | | | | |
| General Electric test reactor | | | | |
| Westinghouse test reactor | | | | |
| Plum Brook reactor facility | | 4 | 0.088 ± 0.014 | 0.086 ± 0.008 |
| | | 4 | 0.170 ± 0.027 | 0.207 ± 0.020 |
| | | 4 | 0.173 ± 0.032 | 0.182 ± 0.017 |
| Air Force nuclear engineering test reactor | Initial | 12 | | 0.0012 |
| Experimental reactors | | | | |
| Experimental boiling-water reactor | | 15 | 0.05 | |
| | | 15 | 0.1 | |
| Vallecitos boiling-water reactor | | 26.5 | | 0.02 |
| | | 12 | | 0.01 |
| Experimental gas-cooled reactor | Initial | 9 | 0.0471 ± 0.0043 | |
| Military reactors | | | | |
| Submarine intermediate reactor, Mark A (West Milton) | | 19-3/4 | 0.1 | |
| Power reactors | | | | |
| Shippingport atomic power station | | | | |
| Dresden nuclear power station | Initial | 29.5 | | 0.0187 ± 0.014 |
| | Retest | 4.33 | | 0.06 |
| Yankee Atomic Electric Co. plant | | 16 | | 0.021 |
| Big Rock Point plant | | 27 | | 0.036 |
| | | 10 | | 0.021 |
| Indian Point plant | Initial | 25 | | 0.014 |
| | Final | 10 | | 0.020 |
| Enrico Fermi atomic power plant | | 30 | | 0.036 |
| Humboldt Bay power plant (dry-well) | Initial | 72 | | 0.025 |
| | Final | 10 | | 0.043 |
| Elk River reactor | Initial | 21 | | 0.05 |
| | Final | 21 | | 0.09 |
| Boiling nuclear superheat (BONUS) reactor | Initial | | | 0.09 |
| | Final | | | 0.12 |
| Carolinas-Virginia tube reactor | | 21 | | 0.074 |
| NS Savannah reactor | Initial | 30 | | 0.3 |
| | Retest | 6 | | 0.108 |
| | Retest | 12 | | 0.31 |
| | Retest | 20 | | 0.465 |
| | Retest | 30 | | 0.86 |
| La Crosse boiling-water reactor | Initial | 52 | | 0.00208 |

[a]These results do not reflect the "as is" leakage rates from these containers, and in some cases the quoted values are beyond the accuracy of the instruments used to make the measurements.

test method used, and, where available, the test results reported. Results of more than one test of a single vessel are shown for a few vessels, representing in most cases the initial test of the bare vessel and the final test after the plant has been completed. It is apparent from this table that the reference vessel method has predominated.

In spite of the greater use of the reference vessel method, there is no general agreement that this should be the preferred method. Several discussions of the relative merits of the absolute and reference vessel methods have appeared in the literature. Although some of the most thorough of these comparisons seem to favor the absolute method, there is not conclusive technical argument that will justify the usage of one more than the other. Perhaps the best discussions of containment vessel leakage-rate testing and comparisons of the two principal methods are presented by Keshock, DeBogdan, Brittan, Jaroschek, and Weippert in refs. 9, 18, 36, 45, and 53.

In ref. 18, Brittan derives the leakage-rate equation for each method and points out the necessity for making more temperature and pressure measurements or more simplifying assumptions for the reference vessel method than for the absolute method. In ref. 53, written specifically in defense of the absolute method in comparison with the reference vessel method, he develops the expressions presented in Section 10.4.12.5 for the possible error in each method.

In ref. 45, Jaroschek and Weippert present a good theoretical discussion of both methods and their relative accuracies. Although they conclude that the absolute method is less complex to use and provides greater measuring accuracy, they were unable to make an experimental comparison of the two methods during a containment vessel leakage-rate test because the reference system did not have sufficient leaktightness. This difficulty may have been reflected in the conclusions of the analytical comparison.

The most recent and most objective and complete theoretical analysis and experimental comparison of the two methods was made in connection with the leakage-rate testing of the Plum Brook reactor containment vessel. This comparison is discussed in ref. 9 and 36. The containment vessel was leakage-rate tested three separate times, using both the absolute and reference vessel methods each time and directly comparing the results of each. The conclusions from these tests are strictly applicable only to the particular test conditions but are significant for a much wider range of conditions. The principal conclusions follow:

1. In all three tests the leakage-rate determinations from both the absolute and reference vessel methods were in substantial agreement.

2. In all cases the reference vessel method consistently yielded results having less scatter than those of the absolute method. This is indicated in Fig. 10.13, which shows the results of the first test.

3. The absolute method offers the advantage of greater overall simplicity, at least for the Plum Brook test conditions. It was felt that for higher test pressures the relative simplicity probably decreased as a result of greater difficulty in making accurate pressure measurements. The test pressure for the tests was 4 psig, a value substantially higher than the 0.3-psig design pressure but considerably lower than required in many tests.

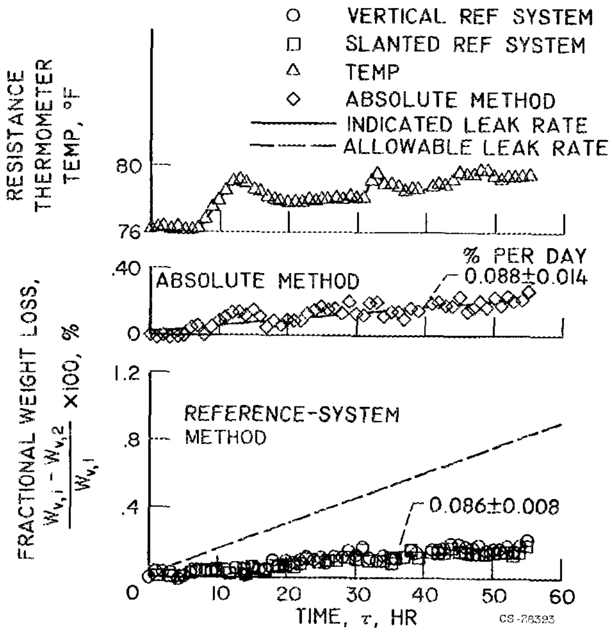


Fig. 10.13. Plum Brook Containment Test Results Comparing the Absolute and Reference-Vessel Methods. (From ref. 9)

Another significant result of the tests and analytical work was the demonstration that the error introduced into the reference vessel method due to thermal lag between the reference system and the containment vessel could be reduced to a negligible value by making the reference vessel diameter sufficiently small (see Sec. 10.4.9.1). An additional advantage cited for the reference vessel method was that it permitted making a rough approximation of the leakage rate while locating and repairing leaks, thus assisting in leak detection and reducing overall testing time.

Of interest in comparing the two methods are the forms of the leakage-rate equations presented in refs. 9 and 36. Equations (10.59) and (10.73) are derived but are then rearranged into the following forms:

1. Absolute Method

$$\frac{w_1 - w_2}{w_1} = 1 - \frac{P_2}{P_1}\frac{T_{s1}}{T_{s2}}\frac{T_1}{T_{s1}}\frac{T_{s2}}{T_2}, \qquad (10.79)$$

where $T_1$ and $T_2$ are the actual average temperatures of the containment air at the beginning and end of the test, and $T_{s1}$ and $T_{s2}$ are the measured values of $T_1$ and $T_2$ but are not identical because of instrument error, personnel error, and inadequate sampling.

2. Reference Vessel Method

$$\frac{w_1 - w_2}{w_1} = 1 - \frac{P_2}{P_1}\frac{(P_1 + \Delta P_1)}{(P_2 + \Delta P_2)}\frac{T_1}{T_1'}\frac{T_2'}{T_1}, \qquad (10.80)$$

where $T_1$ and $T_2$ again are actual average temperatures of the containment air, and $T_1'$ and $T_2'$ are actual temperatures of the reference system air.

As indicated in Section 10.4.9.1, the reference vessel method assumes that $T_1 = T_2'$ and $T_2 = T_2'$, so the temperature ratios in Eq. (10.80) are dropped. But refs. 9 and 36 also show that an equivalent assumption must be made in the absolute method (i.e., that the measured temperature equals the actual average temperature) so that the last two temperature ratios in Eq. (10.79) can be dropped. This then leads to the following two equations:

1. Absolute Method

$$\frac{w_1 - w_2}{w_1} = 1 - \frac{P_2\ T_{S1}}{P_1\ T_{S2}} \ . \tag{10.81}$$

2. Reference Vessel Method

$$\frac{w_1 - w_2}{w_1} = 1 - \frac{P_2\ (P_1 + \Delta P_1)}{P_1\ (P_2 + \Delta P_2)} \ . \tag{10.82}$$

Comparison of these equations indicates that the two methods are exactly equivalent if the reference system is considered to be a particular type of temperature sensor (i.e., a gas thermometer). The selection of one method over the other is then a question of whether a reference system or a system of temperature sensors can better represent the average temperature of the containment air and also which system is more convenient to install and use.

It is apparent that there is no clear advantage for either the absolute or the reference vessel method, and the selection of the method to use for testing a particular containment vessel should be based on the requirements of the particular situation. Undoubtedly the past experience and personal preferences of the individuals involved will enter into the decision to as great an extent as the technical and economic factors.

### 10.4.12 Refinements in Leakage-Rate Determinations

The discussions in Sections 10.4.8 and 10.4.9 covering the absolute and the reference vessel methods of leakage-rate determination follow the measurement and calculational procedures normally employed and correspond to those presented in the proposed ANS standard for containment vessel leakage-rate testing.[1] This section discusses additional refinements to these procedures that have been made in some cases and which may be required if greater accuracy in leakage-rate measurement is desired. It should be cautioned, however, that the use of these refinements alone does not ensure greater leakage-rate accuracy unless the basic measurements of pressure and temperature are sufficiently accurate.

Some of these refinements may even be required to obtain reasonable accuracy in leakage-rate measurement and, therefore, should be considered in every case. This section also gives a brief discussion of the measurement accuracy required to produce a given accuracy in leakage rate and discusses the use of experimental checks to verify leakage-rate measurements and calculations.

### 10.4.12.1 Correction for Changes in Water Vapor

A change in water-vapor content of the containment air during a leakage-rate test would result in an indicated leakage rate either higher or lower than the actual rate. If water condensed out of the air during the test, the indicated leakage would be greater than actual. If water were evaporated into the containment air during the test, the actual leakage rate would be masked to the extent that the increase in water-vapor pressure compensated for a decrease in total containment pressure due to leakage. This latter case could be the more serious, since evaporation, such as from wet concrete or hidden pools of water, is not as easy to detect by visual inspection as is condensation, and failure to correct for it would lead to optimistic results. The proposed ANS containment leakage-rate testing standard specifies that pressurizing for a leakage-rate test be carried out under atmospheric conditions that provide relatively low air humidity, that humidity be monitored during a leakage-rate test, and that precautions be taken to prevent evaporation within the containment vessel during the test, such as covering free water surfaces and draining pools of water. In many leakage-rate tests, water is removed from the pressurized air by cooling or other means before it enters the vessel. It has also been suggested that the air in the vessel be kept saturated during the test, in which case the water-vapor content would be uniquely defined by pressure and temperature so that measurements of water vapor would be unnecessary.[45] However, to improve the accuracy of leakage-rate measurements and to confirm the validity of the test, the change in water-vapor content should be determined accurately during the test in all cases and, if significant, factored into the leakage-rate calculation. Frequently, it is easier to measure and correct for changes in water-vapor content than to take elaborate precautions to minimize these changes.

The error introduced by failing to correct a measured leakage rate for changes in water-vapor pressure is cited in Table 10.9 for several typical conditions. In all cases, a test pressure of 30 psig was assumed. Corrections for change in temperature were made so that the errors indicated are due solely to vapor-pressure changes. The errors shown in Table 10.9 are roughly additive, so, if two or more of the conditions existed during the test, the total error was the sum of the individual errors. Also, if the difference between the initial and final conditions increased, the error was increased in approximately the same proportion. For example, a temperature difference of four degrees would result in an error of 0.1% per day due to a change in water-vapor pressure compared with the 0.05% per day error associated with a two-degree temperature difference. Since the percentage error is inversely proportional to the absolute test pressure, pressures lower than the 30 psig assumed for this

Table 10.9. Possible Errors in Leakage Rates Due to Changes in Water-Vapor Pressure

| Assumed Conditions | Error Introduced by Failure to Make Vapor-Pressure Correction[a] (%/day) |
|---|---|
| 1. Equilibrium between water vapor and excess free water in vessel (100% humidity), $T_1 = 70°F$, $T_2 = 72°F$ | 0.05 |
| 2. Same as 1 except $T_1 = 70°F$, $T_2 = 68°F$ | −0.05 |
| 3. 75% humidity at beginning and end of test, $T_1 = 70°F$, $T_2 = 72°F$ | 0.04 |
| 4. 75% humidity initially, 10 gal of water evaporated into vessel air during test, vessel volume = $10^6$ ft$^3$, $T_1 = T_2 = 70°F$ | 0.06 |
| 5. 100% humidity initially, $T_1 = T_2 = 70°F$, but condensate formed by 5°F drop in temperature during the test is trapped and not returned to equilibrium | −0.13 |

[a]The sign of the error indicates whether it should be added to or subtracted from the uncorrected leakage rate to obtain the actual rate; that is, a negative sign indicates that the actual rate is less than the uncorrected rate.

table would result in correspondingly higher errors. It is evident that errors as great as the leakage rates often specified are possible. Thus the importance of making water-vapor pressure corrections is evident.

The vapor pressure may be incorporated directly into the leakage-rate equations developed in Sections 10.4.8 and 10.4.9 by subtracting the water-vapor pressure from the total pressure wherever it appears. For the absolute method, Eq. (10.60) becomes

$$L_{rc} = \frac{24}{H}\left[1 - \frac{(P_2 - P_{V2})T_1}{(P_1 - P_{V1})T_2}\right]100 , \tag{10.83}$$

where $L_{rc}$ is the percentage leakage rate per day corrected for the effect of water vapor, $P_{V2}$ is the water-vapor pressure at the end of the test, and $P_{V1}$ is the water-vapor pressure at the beginning of the test. For the reference vessel method, Eq. (10.75) becomes

$$L_{rc} = \frac{24}{H} \frac{1}{P_1 - P_{V1}} \left[ \frac{T_1}{T_2} (P_2' - P_{V2}' - P_2 + P_{V2}) - (P_1' - P_{V1}' - P_1 + P_{V1}) \right] 100 \ , \qquad (10.84)$$

or

$$L_{rc} = \frac{24}{H} \frac{1}{P_1 - P_{V1}} \left[ \frac{T_1}{T_2} (\Delta P_2 + P_{V2} - P_{V2}') - \Delta P_1 - (P_{V1} - P_{V1}') \right] 100 \ , \qquad (10.85)$$

where $P_{V2}'$ is the water-vapor pressure in the reference system at the end of the test, and $P_{V1}'$ is the water-vapor pressure in the reference system at the beginning of the test. This equation requires measuring the vapor pressure within the reference system. Since this is usually not convenient, special measures are often taken to maintain a constant water-vapor content in the reference system so that it need not be measured. For example, the reference system may be designed so that it will not collect the water that may condense out of the air during pressurization and the manometer fluid surface, which is exposed to the reference system, may be covered with a low-vapor-pressure liquid. If the water-vapor content of the reference system can be considered constant, the following approximate equation can be used:

$$L_{rc} = \frac{24}{H} \frac{1}{P_1 - P_{V1}} \left[ \frac{\Delta P_2 T_1}{T_2} - \Delta P_1 - \left( P_{V1} - \frac{P_{V2} T_1}{T_2} \right) \right] . \qquad (10.86)$$

The error associated with using this equation instead of Eq. (10.85) is due to the neglect of the term

$$\frac{1}{P_1 - P_{V1}} \left( \frac{P_{V2}' T_1}{T_1} - P_{V1}' \right) ,$$

which should be small compared to the other terms if $P_{V1}'$ approximately equals $P_{V2}'$. Since $P_{V1}$ is always small compared with $P_1$ (less than 1 or 2% in most cases), $P_{V1}$ can be deleted from the denominator of Eq. (10.86) and the following equation can be used with little error:

$$L_{rc} = \frac{24}{H} \frac{1}{P_1} \left[ \frac{\Delta P_2 T_1}{T_2} - \Delta P_1 - \left( P_{V1} - \frac{P_{V2} T_1}{T_2} \right) \right] . \qquad (10.87)$$

As an alternate approach, the apparent leakage rate due to a change in vapor pressure can be calculated separately and then added to or subtracted from the leakage rate calculated with the equations of Sections 10.4.8 and 10.4.9. This approach allows the effect of water vapor to be evaluated quickly and independently so that its significance can be determined before it is incorporated into the total leakage-rate calculation. Following the same development as in Section 10.4.8 and considering that the weight of water vapor is small compared with the weight of air, the apparent leakage rate due to a change in vapor pressure is very nearly

$$L_{rv} = \frac{24}{H} \frac{w_{V1} - w_{V2}}{w_1} 100 \qquad (10.88)$$

or

$$L_{rv} = \frac{24}{H} \frac{1}{P_1} \left(P_{V1} - \frac{P_{V2} T_1}{T_2}\right) 100 \ , \qquad (10.89)$$

where $L_{rv}$ is the apparent percentage leakage rate per day due to a change in vapor pressure, $w_{V1}$ is the weight of the water vapor in the containment vessel at the beginning of the test, and $w_{V2}$ is the weight of the water vapor in the containment vessel at the end of the test. This equation applies using either the absolute or the reference vessel method, if it is assumed, as in Eqs. (10.86) and (10.87), that the air in the reference system has a constant water-vapor content. The corrected leakage rate is obtained by subtracting the apparent leakage rate due to a change in vapor pressure from the leakage rate calculated with either Eq. (10.60) or (10.75), that is,

$$L_{rc} = L_r - L_{rv} \ . \qquad (10.90)$$

Substituting Eqs. (10.75) and (10.89) into Eq. (10.90) results in Eq. (10.87). Substituting Eqs. (10.60) and (10.89) into Eq. (10.90) and rearranging terms gives the following equation, which is equivalent to Eq. (10.83):

$$L_{rc} = \frac{24}{H} \left[1 - \frac{P_2 - (P_{V2} - P_{V1})}{P_1} \frac{T_1}{T_2}\right] 100 \ . \qquad (10.91)$$

This equation was used to correct for changes in water-vapor pressure in the Plum Brook reactor containment vessel leakage-rate test using the absolute method.[36]

A determination of the vapor pressure of water in the containment vessel during a leakage-rate test requires that dew point, absolute humidity, relative humidity, or wet-bulb and dry-bulb temperatures be measured. The vapor pressure can then be obtained in one of the following ways:

1. Dew Point. If the dew point is measured, the water-vapor pressure is simply the saturation pressure of steam at the dew-point temperature and can be read directly from the steam tables.[54]

2. Absolute Humidity. If the absolute humidity (pounds of water per pound of dry air) is know, the water-vapor pressure can be calculated from the following equation based on the definition of vapor pressure:

$$P_v = P \frac{W_v}{W_v + 0.62} , \tag{10.92}$$

where

$W_v$ = pounds of water vapor per pound of dry air (absolute humidity),
$P$ = total pressure of containment vessel,
$0.62 = \frac{18}{29} = \frac{\text{molecular weight of water vapor}}{\text{molecular weight of dry air}}$ .

3. Relative Humidity. If relative humidity is known, the water-vapor pressure can be determined from the definition of relative humidity:

$$\text{R.H.} = \frac{P_v}{P_{sd}} 100 , \tag{10.93}$$

where

R.H. = relative humidity, %,
$P_v$ = vapor pressure of water,
$P_{sd}$ = saturation pressure of water vapor at the dry-bulb temperature.

The value of $P_{sd}$ is obtained directly from the steam tables for the measured temperature. Then, $P_v$ is

$$P_v = \frac{\text{R.H.}}{100} P_{sd} . \tag{10.94}$$

4. Wet-Bulb and Dry-Bulb Temperatures. If wet-bulb and dry-bulb temperatures are measured, a thermodynamic relationship is needed to determine the amount of water vapor in the air. Such a relationship is that given by Eshbach:[55]

$$W_v = \frac{W_{sw} H_{fgw} - 0.241(T_d - T_w)}{H_{fgd} + (T_d - T_w)} , \tag{10.95}$$

where

$W_{sw}$ = weight of saturated vapor per pound of dry air at the wet-bulb temperature,
$T_w$ = wet-bulb temperature,
$T_d$ = dry-bulb temperature,
$H_{fgw}$ = enthalpy of vaporization at $T_w$,
$H_{fgd}$ = enthalpy of vaporization at $T_d$.

The values of $H_{fgw}$ and $H_{fgd}$ are obtained from the steam tables, and $W_{sw}$ is calculated from the following relation, which is a rearrangement of an equation comparable to Eq. (10.92):

$$W_{sw} = \frac{0.62\ P_{sw}}{P - P_{sw}}\ , \qquad (10.96)$$

where $P_{sw}$ is the water-vapor saturation pressure at the wet-bulb temperature and is read from steam tables. The value of $W_v$ calculated from Eqs. (10.95) and (10.96) is used in Eq. (10.92) to obtain $P_v$. It should be pointed out that the psychrometric charts given in various handbooks to determine vapor pressure, humidity, etc. from wet- and dry-bulb temperatures are developed for atmospheric pressure and should not be used in this application because the pressure differs significantly from atmospheric.

Brittan[18] gives expressions similar to those above for calculating the weight of water vapor and the water-vapor pressure from wet- and dry-bulb temperatures. He then uses $P_v$ to calculate the corrected volume of dry air at standard temperature (32°F) and pressure (14.6959 psia).

### 10.4.12.2 Change of Volume

It was assumed throughout the previous sections that the volume, V, of a containment vessel did not change from the beginning to the end of the leakage-rate test and thus V was considered a constant in the derivation of the leakage-rate equations. Actually, there may be a slight change in containment vessel volume due to thermal expansion or contraction or to a change in internal pressure. Brittan[53] gives expressions for both of these errors.

If the effect of thermal expansion is neglected completely and if there are differences between the initial and final temperatures of the test, the percentage error in the percentage leakage rate is given as follows:

$$\%\ \text{error} = \pm 100\ \frac{3\alpha(T_2 - T_1)}{\Delta w/w}\ , \qquad (10.97)$$

where $\alpha$ is the coefficient of thermal expansion and $\Delta w/w$ is the fractional leakage rate. For example, if the temperature at the end of the measurement period is 10°F higher than the initial temperature, an error of 20% in the measured percentage leakage rate would result if a correction were not made (for $\alpha = 6.5 \times 10^{-6}/°F$ and $\Delta w/w = 0.001$). Thus if there is a significant temperature difference between the beginning and the end of the test, expansion or contraction of the vessel could introduce some error.

Brittan shows that the change in shell volume due to a change in internal pressure during a leakage-rate test results in a negligible error in the measured leakage rate. For example, in a typical test of a spherical shell, the error amounts to only 0.2%.

In the leakage rate tests of the Plum Brook reactor containment vessel, which has a large volume of shielding-pool water, corrections were made for changes in the containment vessel volume due to changes in the level of the shielding water caused by leakage through process system lines.[9,36]

When the reference vessel method is used, it also may be necessary to correct for changes in the reference system volume. The percentage change in volume of the reference system can be large if the system volume is small, if the manometer diameter is large, and if the change in manometer fluid level is substantial. Calculations should be performed in each case to determine the significance of this factor. At Plum Brook it was determined that for the 20-ft-long, 2-in.-diam reference vessel used, a change of one vertical inch in the liquid level in the inclined manometer would produce a change in the reference system pressure of 0.092 in. $H_2O$.[36]

### 10.4.12.3 Variation of R

It is generally assumed when deriving the leakage-rate equations using the perfect gas law that R, the perfect gas constant for air, is a true constant. In fact, since air deviates slightly from being a perfect gas, R also varies slightly, and if the conditions at the beginning and end of the leakage-rate test differ significantly, there may be a slight variation in R. The leakage-rate test procedure for the Plum Brook reactor containment vessel[56] considers the variation in R when calculating the inherent uncertainty in the leakage-rate measurement. For the hypothetical example selected, the minimum value of R is 53.840 ft/°F and the maximum value is 53.880 ft/°F, a change of less than 0.1%. Here again, the error is negligible and the variation in R can be neglected.

### 10.4.12.4 Other Corrections

Special circumstances may require other corrections to be made in a particular leakage-rate test. For example, in the leakage-rate tests of the Plum Brook reactor containment vessel, it was necessary to correct for the air introduced into the vessel by the pneumatic controls of the air-conditioning units.[9] This was done by controlling the units with bottled nitrogen during the test so that the amount of nitrogen added to the vessel could be determined by weighing the nitrogen bottle before and after the test. Leakage tests for the Piqua facility were made and reported based on some of these corrections.[57]

10.4.12.5 Error Considerations

The weight of air in the containment vessel is directly proportional to the absolute pressure and inversely proportional to the absolute temperature. Since normal atmospheric temperatures are on the order of 500°R, a 1°F error in temperature is equivalent to a 0.2% error in the weight of the air in the containment vessel. Similarly, if the test is run at an absolute pressure of about 3 atm or about 90 in. Hg, a 0.1-in. error is equivalent to approximately a 0.1% error in weight of the air in the containment vessel. For a low-pressure test, the percentage error in pressure measurement would be even greater: for a 5- to 10-psig leakage-rate test (i.e., about 40 to 50 in. Hg) a 0.1-in.-Hg error is equivalent to a 0.2 to 0.25% error. It is obvious that unless one is willing to take many days to perform the test, instruments with much greater accuracy than 1°F or 0.1 in. Hg are required to measure a containment vessel leakage rate of 0.1% per 24 hr, a rate commonly specified for standard containment vessels. It should be pointed out, however, that the instrumentation need be "reproducibly readable" and not necessarily "absolutely accurate" to the quoted accuracies. This requires good instrumentation with finely calibrated measuring devices but does not necessarily require the instruments to be calibrated against a highly accurate standard.

The proposed ANS standard for containment vessel leakage-rate testing[1] states that instruments for measuring temperature should be accurate to 0.2°F or equivalent and that pressure-measuring instruments should be accurate to 0.1 mm Hg or that both of these measurements should be consistent with the leakage rate specified for the containment vessel. These accuracies are sufficient for the leakage rates specified for most containment vessels. In order to accurately measure a significantly lower leakage rate, improvement in the accuracy of the instrumentation is required or the tests will have to be run for long periods of time. This does not appear possible with standard types of pressure- and temperature-measuring devices.

Brittan[53] presents error analyses for both the absolute and the reference vessel methods. For the absolute method, he gives the maximum possible percentage error in determining the leakage rate as

$$\%\text{ error in leakage rate} = 100\,\frac{2\,\dfrac{e_P}{P} + 2\,\dfrac{e_T}{T}}{\dfrac{\Delta W}{W}}\,, \tag{10.98}$$

where

$e_P$ = error in absolute pressure,
$e_T$ = error in absolute temperature,
$P$ = absolute pressure,
$T$ = absolute temperature.

For the error to be less than 50%, a figure which he considers to represent a valid test, the following expression applies:

$$\frac{e_T}{T} + \frac{e_P}{P} < \frac{1}{4}\frac{\Delta w}{w} . \tag{10.99}$$

For typical test conditions with the absolute test pressure equal to 60 in. Hg, the test temperature about 500°R, and an allowable leakage rate of 0.1% per 24 hr, the allowable error in pressure determination alone for a one-day test, assuming no error in temperature, is

$$\frac{e_P}{P} \leq \frac{1}{4}\frac{\Delta w}{w} = 0.00025$$

or

$$e_P \leq 0.015 \text{ in. Hg} .$$

Similarly, if there is no error in pressure determination, the maximum allowable error in temperature for a one-day test is

$$\frac{e_T}{T} \leq \frac{1}{4}\frac{\Delta w}{w} = 0.00025$$

or

$$e_T \leq 0.125\,°\mathrm{F} .$$

Since actually there will be errors in both pressure and temperature determination, the allowable error for each, if they have equal effect, would be

$$e_P \leq 0.0075 \text{ in. Hg}$$

and

$$e_T \leq 0.063°F \ .$$

In practice, pressure can be measured more accurately than temperature, so allowable errors of 0.0025 in. Hg and 0.104°F might be a more reasonable distribution.

For the reference vessel method, Brittan gives the following expression for the maximum possible percentage error in leakage rate, assuming that all four temperatures (that of the containment air and of the reference vessel air at the beginning and at the end of the test) are measured:

$$\% \text{ error} = 100 \frac{4 \frac{e_T}{T} + 2 \frac{e_P}{P}}{\frac{\Delta w}{w}} \ . \qquad (10.100)$$

In this case, $e_P$ refers to the error in differential pressure. This expression indicates that the possible error due to temperature is twice that for the absolute method because four separate temperatures are measured instead of two. If no temperatures are measured, as is often the case using the reference vessel method, then Brittan states, the error in temperature can be taken to be the uncertainty in knowing whether the temperatures have equalized, since to assure equilization of temperatures, the measurements would have to be made.

In summary, Brittan suggests that the required accuracy of pressure and temperature determinations be dependent on the magnitude of the specified maximum allowable leakage rate and that the total possible error be less than 50% of the allowable leakage rate. He thus proposes the following allowable leakage rates for the absolute and the reference vessel methods:

1. <u>Absolute Method</u>

$$\frac{e_P}{P} + \frac{e_T}{T} < 1/4 \text{ of the allowable fractional leakage during the test period.} \qquad (10.101)$$

This also may be expressed as

$$\text{Allowable leakage rate} \geq \frac{4}{t} \left( \frac{e_P}{P} + \frac{e_T}{T} \right) 100 \ , \qquad (10.102)$$

where t is the length of the test period in days.

2. Reference Vessel Method

$$\frac{e_P}{P} + \frac{2e_T}{T} < 1/4 \text{ of the allowable fractional leakage during the test period,} \qquad (10.103)$$

or

$$\text{Allowable leakage rate} \geqslant \frac{4}{t}\left(\frac{e_P}{P} + 2\,\frac{e_T}{T}\right) 100 \,. \qquad (10.104)$$

The "second power equation," or "error propagation law," has also been used to estimate the uncertainty or possible error in the calculated leakage rate due to the uncertainties in the measured variables upon which the leakage rate depends. This equation may be expressed as follows:

$$e_L = \left[\left(\frac{\partial L}{\partial V_1}\, e_1\right)^2 + \left(\frac{\partial L}{\partial V_2}\, e_2\right)^2 + \ldots + \left(\frac{\partial L}{\partial V_n}\, e_n\right)^2\right]^{1/2} , \qquad (10.105)$$

where

$L$ = result of the calculation, in this case the leakage rate,
$e_L$ = uncertainty interval of the result,
$V_1, V_2, \ldots V_n$ = independent measured variables,
$e_1, e_2, \ldots e_n$ = uncertainty interval for each of the measured variables.

In their analysis of the leakage-rate tests of the Dido and Merlin reactor containment vessels, Jaroschek and Weippert[45] applied this equation to both the absolute and reference vessel methods. For the absolute method the uncertainty or, as they term it, the measuring tolerance in leakage rate is shown to be:

$$e_L \approx \pm \frac{1}{\Delta t}\left[2\left(\frac{e_P}{P}\right)^2 + 2\left(\frac{e_T}{T}\right)^2 + 2\left(\frac{e_X}{X^*}\right)^2\right]^{1/2} , \qquad (10.106)$$

where

$\Delta t$ = interval of test,
$P$ = absolute pressure,
$T$ = absolute temperature,
$X^*$ = absolute humidity + 0.622,
$e_P$, $e_T$, $e_X$ = uncertainties, or measuring tolerances, in pressure, temperature, and absolute humidity, respectively.

The corresponding equation for the reference vessel method is given as:

$$e_L \approx \pm \frac{1}{\Delta t} \left[ 4 \left( \frac{e_T}{T} \right)^2 + 2 \left( \frac{e_X}{X^*} \right)^2 \right]^{1/2} . \qquad (10.107)$$

Pressure was not included in Eq. (10.107) because $e_P/P$ in the reference vessel method is small compared with the other measuring tolerances. For the Dido and Merlin test conditions and assumed measuring tolerances, the above relationships indicated the absolute method to be more accurate.

Equation (10.106) was also used in the procedure for the leakage-rate test of the Plum Brook reactor containment vessel.[56] Using a hypothetical example, the total uncertainty in leakage rate was about ±11% and it was concluded that the greatest part of this uncertainty was due to the uncertainty in measurement of the dew point.

Keshock and DeBogdan[36] point out in their comparison of the absolute and the reference vessel methods that the effect on leakage rate of an error in temperature measurement is much less with the reference vessel method than with the absolute method. They state that since in the reference vessel method the temperature can be considered to be represented by the value of $P + \Delta P$, the measurements of temperature and pressure are, in a sense, coupled, and pressure-measurement errors will be largely cancelled out. For example, with values typical of the Plum Brook reactor containment tests, a 1% error in $P_2$ would result in an error of only 0.01% in the ratio $P_2/(P_2 + \Delta P_2)$. On the other hand, with the absolute method, temperature and pressure measurement errors are independent so that a 1% error in $P_2$ would produce a 1% error in the ratio $P_2/T_2$. This was illustrated by the results of one of the Plum Brook leakage-rate tests. In this test, an apparent pressure measurement error at a few data points produced no noticeable error in the leakage rate determined with the reference vessel method but caused the leakage rate determined with the absolute method to fall off the scale at these points.

#### 10.4.12.6 Experimental Checks on Leakage-Rate Determinations

Experimental checks in support of leakage-rate determinations may take various forms, ranging from experimental verification of the accuracy of the instrumentation used, to the simultaneous conduct of tests with different testing methods, to the complete rerun of a particular test. Numerous data points taken throughout a test tend to establish trends and indicate the scatter of the data being accumulated. Excessive scatter caused by known and recurring phenomena may be cause for extending the time period of the test in order to obtain statistical results.

One method used in a few containment vessel leakage-rate tests has been to record the necessary data for making absolute leakage-rate calculations at the same time that the data for the reference vessel leakage-rate calculations are being observed. Another method, successfully demonstrated in the leakage-rate tests of the Plum Brook and NS Savannah containment systems, is to superimpose a known leakage rate upon the existing leakage rate during the latter part of the test. The degree to which the increase in the observed leakage rate equals the additional known leakage rate will then provide an additional basis for determining the validity of the test. In the Plum Brook test the known leakage rate was established by bleeding air from the vessel through a gas flowmeter at a rate roughly equal to the allowable leakage rate.[36] This was done during the last 12 hr of a 60-hr test period. Figure 10.14 presents the resulting data for the imposed controlled leakage rate of 0.226 ± 0.002% per day. Subtracting this value from the indicated rates for the absolute and reference methods gave the following respective leakage rates: 0.198 ± 0.055 and 0.164 ± 0.040% per day. These rates compared with preceding rates for both methods are seen to overlap on Fig. 10.14. The controlled leakage rate therefore substantiated the general agreement and the accuracy of both measuring methods.

### 10.4.13 Special Considerations for Various Types of Containment Structures

The leakage-rate testing methods outlined in preceding portions of this section reflect the practices used and the experience gained to date in containment vessel leakage-rate testing. Since nearly all of this experience has been with steel-shell containment vessels, the methods presented were developed primarily with this type of containment in mind. In most cases these methods are applicable to other types of containment structures, but further consideration may be required before they are applied directly. The following paragraphs briefly outline some of the special considerations which may be involved in the leakage-rate testing of various types of containment structures. For any particular containment design a much more thorough analysis of the factors involved will be required.

#### 10.4.13.1 Conventional Buildings

Structures similar to conventional buildings are used for reactor containment when a reactor accident will not produce a substantial pressure rise and when a high degree of leaktightness is not required. Containment structures of this type are often operated at reduced pressure; that is, leakage from the building is prevented by maintaining a ventilation system flow rate sufficient to produce a slightly negative pressure within the building so that all leakage is inward. The ventilation exhaust is usually directed up a stack, with provision for filtering if necessary. For buildings operated in this way, it may be that the only leaktightness requirement is that the specified reduced pressure be maintained with a given ventilation blower capacity. In this case, a leakage test would consist only of measuring the differential pressure of the building with a water manometer while the ventilation system was operating.

If a maximum leakage rate is specified for the building, in addition to a reduced pressure, a leakage-rate test can be performed by measuring the flow rates of the ventilation system intake and exhaust with conventional gas flowmeters. The leakage rate is then the difference between these two flow rates. This is a convenient and simple means of determining the leakage rate with reasonable accuracy, since all leakage is channeled in one flow path. Leakage rates of conventional buildings are

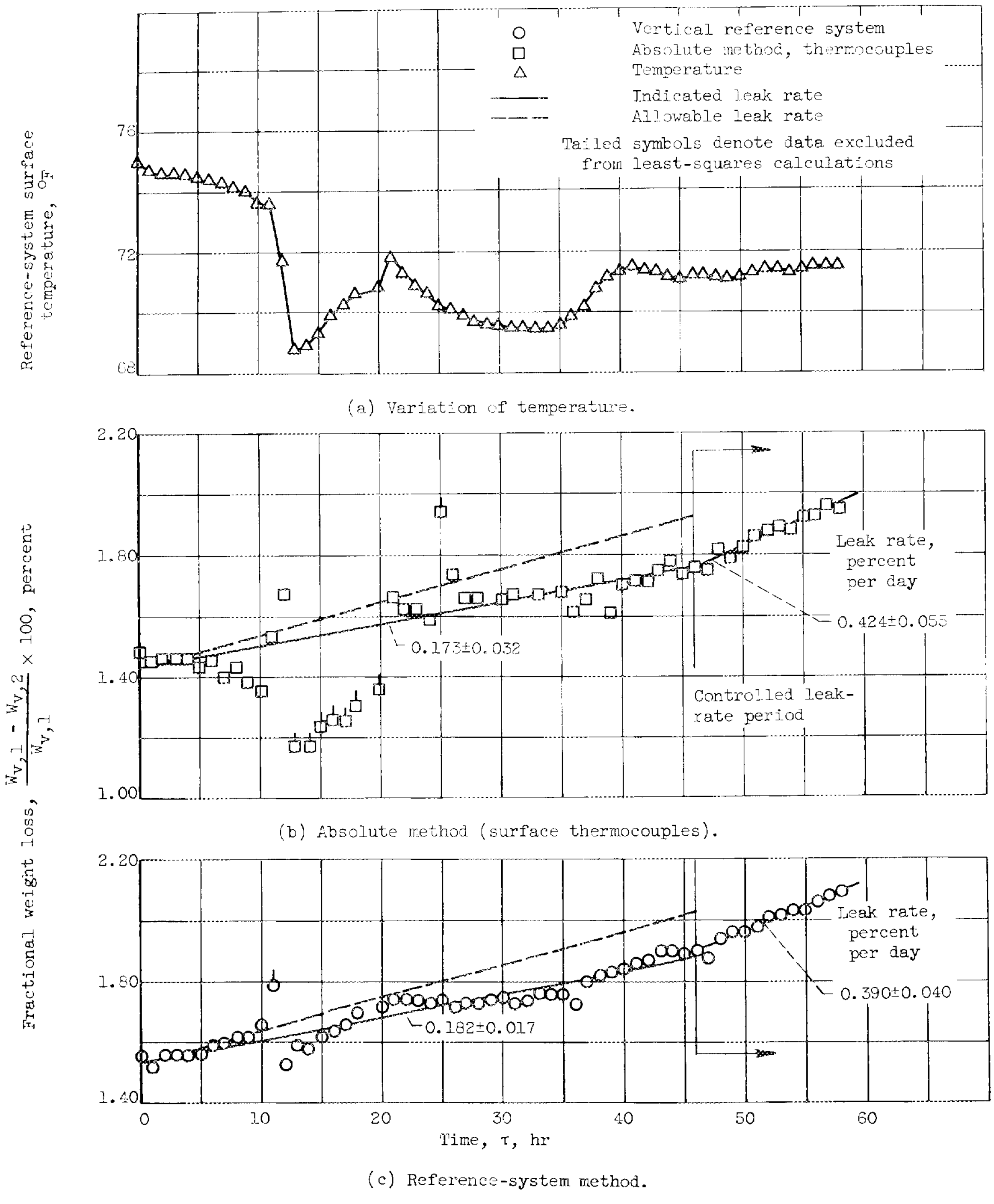


Fig. 10.14. Controlled Leakage Rate Data Compared with Absolute and Reference System Data Obtained for Plum Brook Containment Vessel. (From ref. 36)

usually large enough — from 100% per day to 100% per hour or more — to be measured in this way without special techniques or long test periods. If special provisions are taken to minimize leakage, such as using special doors, joints, seals, coatings, etc., leakage rates as low as a few percent per day, may be achievable.[13] In this case, it may be necessary to use more sensitive devices to measure the leakage rate. However, if the building is to be operated at reduced pressure, the same general procedure of measuring the ventilation system flow rate is usually used.

If the building is designed for a slight positive pressure and a maximum allowable leakage rate is specified, one of the pressure-decay leakage-rate-testing methods should be used. The discussion in the following paragraph on low-pressure containment applies to this case. Generally, leakage-rate testing should be performed with the leakage flow in the same direction as it would be under accident conditions. Thus, if the containment structure is designed for a positive pressure during an accident, the leakage rate should be measured in the same way, since the leakage rate determined in a reduced-pressure test might be substantially different from that measured in a pressure-decay test. Extrapolation would be required, as discussed in Section 10.4.6. If the positive pressure of a retest was sufficiently low that it might be masked by test conditions, such as wind pressure external to the containment, precautions would have to be taken to avoid such conditions in order not to invalidate the test.

### 10.4.13.2 Low-Pressure Containment

Containment vessels with low design pressures (5 psig and less) and low maximum allowable leakage rates are tested in the same way as conventional containment vessels with higher design pressures. The leakage-rate testing methods described in previous sections of this chapter are used. However, since the sensitivity of the leakage-rate test decreases with decreasing test pressure, a longer test period may be required in order for a low leakage rate to be measured with sufficient accuracy. The test may be accelerated if it is possible to perform the test at a pressure higher than that for which the leakage rate is specified. This was done in the leakage-rate tests of the Plum Brook reactor containment vessel. In this case the maximum accident pressure was 0.3 psig, but the leakage-rate tests were conducted at 4 psig.[36] This is not normally possible, however, since the maximum allowable leakage rate is usually specified at the maximum accident pressure, which is usually the vessel design pressure and close to the maximum the vessel can safely withstand.

For low-pressure containment vessels where very low leakage rates are not required, the same general testing methods would probably be used, but there would be less need for an extended test period to obtain the required measurement accuracy.

### 10.4.13.3 Pressure-Relief Containment

The type of containment system that vents off the initial pressure surge during an accident and then is closed up before the fission products are released from the core is tested much like a low-pressure containment vessel. The main difference is the isolation system, which must operate reliably and have the necessary degree of leaktightness after closing. This type of containment probably will not be used where a high degree of leaktightness is required, so it may not be necessary to be able to measure very low leakage rates.

#### 10.4.13.4 Pressure-Suppression Containment

A pressure-suppression system may be more difficult to test than a conventional containment vessel, since it consists of two or more interconnected vessels that may have different design pressures and different maximum allowable leakage rates. During construction the vessels may be isolated for pressure testing, and leakage-rate tests can then be performed using a pressure-decay method. However, after the system has been completed, the chambers cannot normally be isolated from each other except by the water in the suppression pool covering the ends of the connecting piping. A leakage-rate test may then have to be run at the same pressure in each chamber and the measured leakage rate extrapolated to different pressures if necessary.

In a pressure suppression system the suppression chamber will generally have a lower design pressure, a smaller total volume, or both, than a comparable conventional containment system. The sensitivity of the leakage-rate test will thus be somewhat less. More temperature and pressure instrumentation may be required to accurately represent the air in the various parts of the system, and, because some parts of the system may not be readily accessible, the test instrumentation may be more difficult to install. Humidity measurements would be more critical because of the large exposed surface of water. Because of these difficulties, special systems and techniques may have to be devised to demonstrate the leaktightness of the overall system. A makeup-air method has been proposed for checking the leaktightness of the Humboldt Bay pressure-suppression system (see Sec. 10.4.10.1).

#### 10.4.13.5 Underground Containment

Underground containment structures would be leakage-rate tested using the pressure-decay test methods used for comparable aboveground structures. Temperature variations should be much less because of the insulating effect of the surrounding earth, so fewer temperature sensors or reference vessels should be required and there should be less scatter in the test data. Humidity measurements would be particularly important because of the possible inleakage of ground water.

Underground structures designed so that the external ground-water pressure exceeded the maximum internal pressure would be inherently quite leaktight and might be used where a particularly high degree of leaktightness was required. In this case, there would be great incentive to perform a leakage-rate test of even greater than normal sensitivity in order to demonstrate this leaktightness.

Detection and location of leaks in the vessel wall would be difficult because the outside surface would not be accessible for soap bubble testing. However, access openings to the vessel and electrical and piping penetrations probably would all be located in the same accessible area so that these areas of greatest potential leakage could be tested without great difficulty.

### 10.4.13.6 Multiple-Barrier Containment

Since little experience can be cited for multiple-barrier containment, the discussion in this section must of necessity be somewhat speculative. The envelopes around the reactor vessel provided by a multiple-barrier containment system allow considerable flexibility in leakage-rate testing and offer the possibility of using some different methods. Leakage from the primary vessel can be channeled into one flow path in the annular space between the inner and outer barriers so that conventional gas flowmeters can be used to measure the leakage. For this test the annular space should be kept at atmospheric pressure to minimize inward leakage through the outer barrier. In an attempt to obtain greater sensitivity, an adulterant gas can be added to the air within the primary vessel, and its concentration can be measured in the annular space. Helium can be used for this purpose and its concentration measured very accurately with a mass spectrometer (see Sec. 10.4.10.2). Of course, the pressure-decay method of leakage-rate testing can also be used.

Leakage-rate testing of the outer containment barrier depends on the system design and on the degree of leaktightness required of the outer barrier. If, as is usually the case, the annular space is kept at reduced pressure so that leakage is inward, the leakage rate can be determined by measuring the flow rate of air out of the annular space. If the annular space could reach a positive pressure so that leakage was outward, a pressure-decay leakage-rate test could be performed. In both cases, since the leakage rate of the inner barrier would be measured along with that of the outer barrier, the inner-barrier leakage rate would have to be determined independently and subtracted from the total or assumed to be negligible. In systems where the inner barrier had much greater leaktightness than the outer barrier, such as for a steel shell enclosed by a conventional building, the inner-barrier leakage rate could be neglected in determining the outer-barrier leakage rate. Where the leakage rates of both barriers were approximately the same, the inner vessel could be kept at the same pressure as the annular space during the test so that it would have negligible leakage.

## 10.5 A TYPICAL INITIAL INTEGRITY TESTING PROCEDURE

This section presents a step-by-step procedure for a typical initial integrity test consisting of a strength test, local leak tests, and an integrated leakage-rate test for a typical steel-shell containment system. The purpose of this section is to indicate the continuity that normally exists in performing these tests and, in a sense, to provide a summary of Sections 10.2, 10.3, and 10.4. This procedure should be considered as only typical and not a mandatory or necessarily a recommended procedure. The use of the reference vessel method for integrated leakage-rate testing in this procedure should not imply that this method is considered preferable to the absolute method, but rather, it reflects the greater use of this method for tests of containment vessels to date. This step-by-step procedure is generalized and intended to be as widely applicable as possible. Thus it does not contain the details required to perform a proof test of a specific containment vessel. In addition steps have been incorporated that may not be necessary for all circumstances, for example, humidity measurements and the use of blowers and strain gages.

A schematic diagram of the equipment and instrumentation required for this typical procedure is presented in Fig. 10.15.

## Part A. Preliminary

1. Blank off all penetrations except for the personnel and equipment air locks and the penetrations to be used for the test.
2. Install reference vessels at predetermined locations within the containment vessel and install all necessary connecting tubing and valves. Install a differential pressure manometer outside the containment vessel and connect tubing between the reference system, the manometer, and the containment vessel.
3. Test the reference system, including the vessels, tubing, and valves, for leaktightness using a halogen or helium leak detection test.
4. Install piping and valves between
    a. The containment vessel and the pressure gage,
    b. The containment vessel and the compressors (note that an adequately sized pressure-relief system is required in accordance with the proposed ANS Standard; this pressure-relief system may be installed directly on the containment vessel or, if the lines are adequately sized, may be a part of the compressor system),
    c. The air locks and the compressors.
5. Install temperature measuring instruments at previously determined locations within the containment vessel.
6. Connect pressure gage lines to the pressure gages and pressure recorders at the remote test control point. Connect the temperature instrument leads to temperature recorders at the control point.
7. Install dew-point instruments within the containment vessel.
8. Install temporary blowers within the vessel that are capable of operating continuously at vessel design pressure. These blowers are to be used to circulate the air during the leakage-rate test.
9. Close the doors. If air locks are used, close the inner doors of air locks and leave the outer doors open. Special care should be taken that these operations are properly performed in order to prevent the doors from blowing open when the vessel is pressurized.
10. Conduct a complete visual inspection of the containment vessel and all penetrations to determine that all openings have been sealed. Correct any obvious sources of leakage.

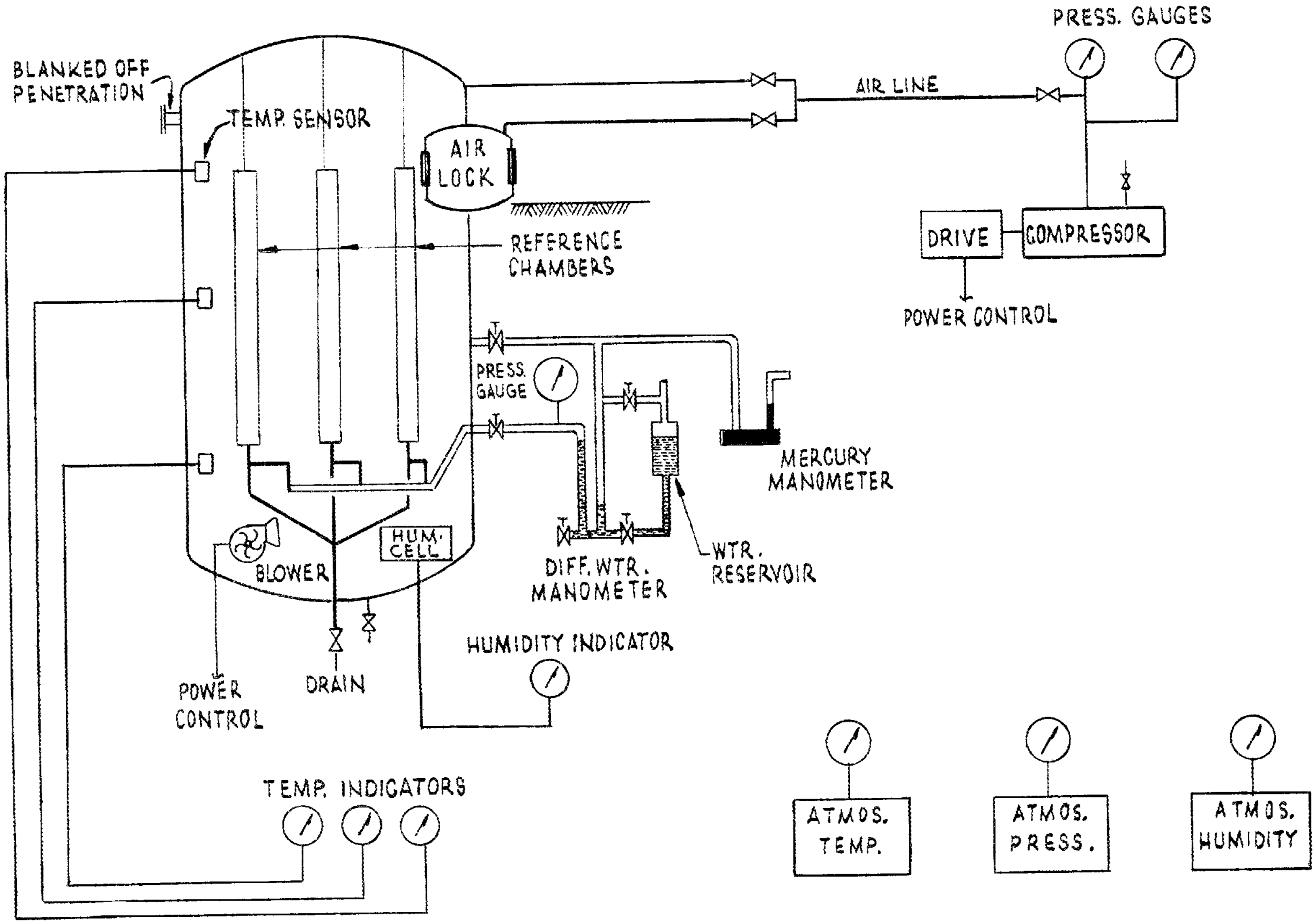


Fig. 10.15. Schematic Diagram of Equipment Required for a Typical Containment Integrity Test.

## Part B. Strength Test

11. Start the air compressor and begin pressurizing the containment vessel. During the initial part of the pressurization, personnel should be stationed around the containment vessel to look for obvious leaks and to determine that penetration seals are seating properly.

12. Continue operating the compressor until the containment vessel has been pressurized to a small fraction of design pressure, often to approximately 5 psig and then stop the compressor.

13. Conduct a soap bubble test of the containment vessel shell, except for the air locks, which are not pressurized, giving special attention to the shell seams, gaskets of manholes and doors, and temporary test covers.

14. If air locks are used, close the outer doors of the air locks and pressurize the air locks to a small fraction of design pressure, often approximately 5 psig.

15. Conduct a soap bubble test of the outer doors and seams of the air locks that were not previously tested.

16. Repair or seal any leaks detected by the soap bubble test. Two types of leaks requiring different corrective action may be found during the soap bubble test. These are

a. Large fissures or cracks that might affect the basic integrity of the vessel. If such imperfections exist, further pressurization might have serious consequences; therefore, the vessel must be depressurized and the imperfection repaired before the vessel may be pressurized further.
b. Minor leaks that do not affect structural integrity but which could affect the vessel leakage rate. These leaks may be sealed temporarily, providing it is determined safe to do so, and permanent repairs may be made later. The vessel would need to be depressurized only if it were necessary to seal the leak.

17. Close the air supply line to the air locks and open the air lock blow-off valves to release the pressure in the air locks.

18. Restart the compressor and continue pressurizing the containment vessel while observing the following clearance rules:

a. All unauthorized personnel must maintain a clearance of 1200 ft from the containment vessel while pressure is being increased above 5 psig and until the pressure test and final soap bubble test have been successfully completed.
b. Authorized personnel who are conducting the test must maintain a clearance of 600 ft from the outside of the vessel, except during strain gage readings and soap bubble testing at designated pressures.
c. During the leakage-rate test (Part C) only authorized personnel shall be allowed on or adjacent to the vessel and instruments. No work shall be permitted within 25 ft of instruments, valves, and the containment shell.

19. Pressurize the containment vessel to approximately one-half the vessel test pressure.

20. While holding at one-half pressure, visually check major penetration seals. If the pressure presents a great hazard, then the pressure should be reduced before proceeding with visual inspection.

21. Increase pressure to full test pressure in increments of approximately one-tenth the test pressure.

22. Hold at vessel test pressure approximately 20 min.

23. Pressurize the air locks to full test pressure to test the outer air-lock doors.

24. Hold at full test pressure for another 40 min, adding or releasing air to compensate for temperature variations if necessary.

25. Reduce pressure in the containment vessel and air locks to design pressure.

26. Conduct a soap bubble test of the containment vessel, including the outer doors and seams of the air locks, all seams of the shell and nozzles, all gaskets of manholes, and all penetration test covers.

27. If any leaks are found, use the following procedure:

a. Immediately repair any leaks of sufficient magnitude to indicate that the structural integrity of the vessel might be affected. The vessel should then be retested at full test pressure, and the repaired area retested with a soap bubble test at design pressure.

b. A leak considered not to affect the structural integrity of the vessel but which might prevent a successful leakage-rate test may be temporarily sealed, if possible, and the test continued. After the leakage-rate test, final repairs should be made and necessary local leak testing of these repairs should be conducted.

c. A leak considered not to affect the structural integrity of the vessel but which cannot be sealed and which would prevent a successful leakage-rate test may be repaired, depressurizing during repairs if necessary, or the structural test may proceed and the repairs be made before conducting the leakage-rate test.

28. Open the air-lock blowdown valves and blow down the air locks to atmospheric pressure while maintaining pressure in the containment vessel. Open the air-lock outer doors.

29. Conduct a soap bubble test of the air-lock inner doors and seams not previously checked at design pressure.

## Part C. Leakage Rate Test

30. Allow equilibrium conditions to be established within the containment vessel during the period of uniform atmospheric conditions between midnight and dawn. Before starting the leakage-rate test, the containment vessel air temperature should have stabilized so that it does not vary by more than a few degrees over a period of several hours.

31. Drain off any condensate from the reference system and the containment vessel.

32. Check auxiliary pressurized systems within the containment envelope to ascertain that no leakage to the containment is occurring.

33. Start the temporarily installed blowers to circulate the containment vessel air.

34. Fill the differential pressure manometer with water.

35. Open the containment vessel blowoff valve and reduce the containment pressure slightly to produce a small differential pressure on the water manometer to ensure that the manometer is functioning properly. Alternatively, the reference system may be established initially at a slightly reduced pressure by bleeding air out of the reference system or by restarting the compressor and increasing pressure in the containment vessel.

36. Record the following data hourly, or more frequently, throughout the leakage rate test:

a. Atmospheric temperature,
b. Atmospheric barometric pressure,
c. Containment vessel gage pressure,
d. Containment vessel absolute pressure as indicated by the sum of b and c,
e. Containment vessel average temperature as indicated by the average of all containment vessel temperature-measuring instruments,
f. Differential pressure between the containment vessel and the reference system as measured by the water manometer,
g. Dew point of the containment vessel air.

37. Plot items a, d, e, and f of step 36 against time.

38. Continue the test for at least 24 hr until stable atmospheric conditions, comparable to those at the beginning of the test, are reached, that is, until a succeeding midnight to dawn period. The length of the test period should be approximately a multiple of 24 hr. The times selected for the beginning and the end of the test period should be those when the differential pressure and the containment vessel temperatures are the most stable, that is, when they have been changing the least with time. The initial and final temperatures should also be as nearly the same as possible, but it is more important that they be stable.

39. Calculate the containment vessel percentage leakage rate using Eq. (10.75) of Section 10.4.9. If necessary, correct for a change in water-vapor content of the containment vessel air using the equations of Section 10.4.12.1.

40. If the calculated percentage leakage rate is acceptable, release the containment vessel pressure to atmospheric pressure and proceed to Step 43.

41. If the calculated percentage leakage rate is slight but there is doubt that the results are a fair indication of the actual leakage rate, continue the test for an additional period of time.

42. If the calculated percentage leakage is substantial, recheck the containment vessel, the reference system, connections, valves, and instruments for sources of leakage, and repeat the leakage-rate test if necessary.

43. Enter the containment vessel and make a thorough visual inspection for any indication of effects of the pressure test or for any other abnormalities that might have affected test results.

(Although not a part of typical current practice, leakage tests at various pressures up to design pressure would be desirable in order to permit the extrapolation of leakage rates subsequently obtained at lower pressures.)

## 10.6 RETESTING

### 10.6.1 Requirements for Retesting

Because of the importance of high containment integrity in reducing the hazards associated with a nuclear power plant and because of the possibility that deterioration of certain parts of the containment system may occur over long periods of time, it is necessary that periodic inspection and retesting be performed to ensure that containment integrity is being maintained. It is generally agreed that retests for strength are not required unless additions or modifications to the vessel are made, since any deterioration that might be expected to occur would not significantly affect structural integrity. However, deterioration of seals to the extent that the leakage rate is substantially increased is not unlikely, and some means of demonstrating leaktightness is required. There are no generally applicable requirements for integrated leakage-rate retesting, but some plants have had retesting requirements established on an individual basis. As larger plants are built closer to populated areas and greater dependence is placed on containment integrity, the current trend toward integrated leakage-rate retesting can be expected to increase. Since a containment structure is essentially a fluid-mechanical system, periodic tests do not seem unreasonable in view of present code requirements for other similar systems. However, even periodic tests at full pressure may not be sufficient to demonstrate container integrity, particularly if significant fluctuations occur. Consequently a more thorough approach would be to design individual penetrations for continuous monitoring at up to design pressure.

Section 17, Periodic Inspection and Testing, of the proposed ASA standard for steel containment structures (Appendix E) contains the following provisions regarding periodic inspection and retests:

> "1701 – The purpose of periodic inspection and testing is to insure that the allowable leakage rate is not exceeded during the life of the plant.
>
> The pressure used for periodic leakage testing shall be as close as possible to that used for the Initial Leak-Rate Test but it is often limited by the fact that is is difficult to subject the final installation to full pressure tests because some instrumentation and equipment might suffer.
>
> "1702 – Most leakage will normally occur through large access openings, air locks, valves, etc., which are subject to use during normal operations, and it is recommended that these be tested at least once a year. It is recommended that other removable covers be tested after each period of use in which they are removed.

Integrated leakage rate retest shall be required only after significant repairs on the containment structure have been made, or if excessive corrosion or other deteriorative processes are evident from the annual inspection. Such a test shall be performed at least once within the first five years of reactor operation.

"1703 - The continuous structural integrity and leak-tightness might be jeopardized by the following conditions which should be subject to an annual inspection:

(a) Unequal settlement of the foundations.
(b) Corrosion.
(c) Deterioration with consequent leakage at a connection, door, or removable cover.
(d) New work on the containment shell.
(e) Mechanical impact damage.
(f) Cracking at points of stress concentrations."

The proposed ANS standard for containment vessel leakage-rate testing[1] contains the following provisions regarding reinspection and retesting:

"8.1 Reinspection. Annual reinspection is recommended to determine whether visual evidence of deterioration of the structure has occurred and whether this might affect its tightness with respect to the leakage rate. Such inspection should include evidence of unequal settlement of the foundations, significant corrosion, significant weathering of sealing compounds or other nonmetallic materials, cracking at weld areas or other regions of stress concentration and damage resulting from operations or accidents. Penetrations and closures should be examined and their functional reliability determined.

"8.2 Local Leak Detection Retests. Localized pressure tests, such as those described in 4.4, should be made whenever annual inspection tests or other circumstances show deterioration or otherwise indicate the desirability of such retests. Localized pressure tests shall be made whenever repairs or new construction are involved. A record of local leak test results should be maintained for reference."

Since the presently proposed standards are still in a development stage, a firm guide to retesting frequency may not be available until industry develops testing practices and acquires experience to justify the evolvement of standards.

### 10.6.2 Limitations

A pressure test or a full-pressure integrated leakage-rate test of a containment vessel containing a completed plant may be very difficult to perform. Much of the shell surface is inaccessible so that it could not be properly inspected after a pressure test. In many cases, some instrumentation and equipment installed within the containment vessel of a completed plant might be damaged if subjected to design pressure.

To avoid the possibility of damaging certain instrumentation and equipment or to avoid having to remove it, leakage-rate retests may be conducted at reduced pressures, often at 5 psig or less. The measured leakage rate at the reduced pressure must then be extrapolated to an equivalent leakage rate at design pressure. This extrapolation introduces several uncertainties, as discussed in Section 10.4.6. For example, it is not certain that all the leaks which might occur at design pressure will occur at the reduced pressure. Also, there is no general agreement on the type of flow through the leaks, and, therefore, there is no generally accepted method for extrapolating leakage rates to higher pressures. Furthermore, the sensitivity of the leakage-rate test varies directly with testing time and inversely with test pressure so that for an equivalent sensitivity, a reduced-pressure test would be proportionately longer. It is thus apparent that a reduced-pressure leakage-rate test will be less accurate than a similar test at design pressure.

An additional limiting factor in retesting of vessel leakage rate is the time required to conduct such a test. To obtain sufficient accuracy, it is likely that the test will have to run over a period of two or more days, during which time access to the vessel for operation or for maintenance work is not permitted. For a commercially operating power plant, this loss of two or more days of operation could impose a substantial economic penalty. Economic pressures will undoubtedly, therefore, foster the development of test procedures that will give test results of adequate and unquestioned confidence coupled with an economically reasonable incurred cost. The design of a containment system for complete continuous monitoring would certainly provide the desired leakage data, but the cost over the long term has not been established.

### 10.6.3 Methods

Both the absolute method discussed in Section 10.4.8 and the reference vessel method discussed in Section 10.4.9 could be used for the reduced-pressure tests without modification, except that the testing period might be longer to provide adequate sensitivity, and more temperature-measuring instruments or reference vessels might be required to give adequate temperature compensation.

Some of the other methods discussed in Section 10.4.10 may be more attractive for retesting than the absolute or reference vessel methods because of the pressure and accessibility limitations. In particular, the method of checking individual penetrations has several advantages. This method, in which an airtight box is built around each penetration that might be expected to leak, allows these penetrations to be tested at full design pressure without pressurizing the entire containment vessel. A high degree of accuracy can be achieved with this method provided all sources of leakage are known. This method can be used to test leakage of these penetrations at any time without interrupting operation or maintenance of the plant and can even be used for continuous monitoring. Other local leak testing methods can be used to check for leakage at individual locations (see Sec. 10.3).

Continuous monitoring of the differential pressure between the containment vessel and a reference system is an alternative to retesting that can be used on those plants where no access is permitted to the containment vessel during operation or limited access, with special provisions to prevent false indications, and where a small positive pressure is maintained within the vessel. This method was planned for use on the Yankee plant.[51] Even though the vessel pressure during operation is low, continuous monitoring will indicate trends in leakage rate and will provide a good indication of containment integrity. At Yankee, further indication of the leakage rate can be provided by measuring the amount of air that must be added to the containment vessel to bring the differential pressure back to its initial value.

The current trend is toward requiring infrequent integrated leakage-rate retesting of containment vessels, perhaps in addition to somewhat more frequent testing of penetrations. The pressure at which such a test is to be conducted is a major difficulty in attempting to reach agreement on an acceptable retest procedure. Three alternatives are possible:

1. The test could be conducted at full design pressure. This would give the most accurate results and would be the most representative of an accident condition but, as indicated above, it would be frequently very difficult or impossible because of the danger of damaging certain equipment and instrumentation. If such retesting were considered during design of the plant, it might be possible to use only equipment that could withstand full design pressure.

2. A reduced-pressure test could be performed and the results extrapolated to design pressure based on a curve of leakage rate versus test pressure previously determined in a series of leakage-rate tests performed on the vessel at various pressures. This could be nearly as accurate as a full-pressure test, but the series of tests required to establish the curve would require considerable testing time. Moreover, for the curve to be valid, the series of tests would have to be run with the vessel in its final configuration, at which time it may be as difficult to perform a full-pressure test as it would be for a retest.

3. A reduced-pressure test could be performed and the results extrapolated to design pressure using a generalized, but conservative, leakage-rate versus pressure relationship. If the measured leakage rate is well below the allowable rate, this method would undoubtedly be preferred. Of course, agreement would have to be reached on what constituted a "conservative" relationship.

### 10.6.4 Experience

Only limited results of leakage-rate retests have been published, since only a few plants have been in operation long enough to have required retests. At Dresden, a leakage-rate retest was conducted at a test pressure of 4.33 psig, as compared with the vessel design pressure of 29.5 psig.[58] The test pressure was limited because of possible damage to the GM tubes at pressures greater than 5 psig. The measured leakage rate was extrapolated to an equivalent leakage rate at design pressure using the orifice equation. An extrapolation factor of 1.155 was obtained.

The ventilation system butterfly valves and the personnel and equipment air-lock doors and equalizing valves were the principal, if not the entire, sources of leakage during this test. The reference vessel method was used for this retest, as it was for the initial leakage rate test, with 20 reference vessels distributed around the inside of the containment vessel. The reference system was checked prior to the test with a freon leak test and a pressure test.

Leakage-rate retests of the air locks and of the ventilation system butterfly valves were conducted at Shippingport.[32,33] These tests indicated that significant amounts of leakage occurred around the air-lock seals and around the butterfly valves but that after adjustment and repair an acceptable leakage rate was achieved.

Several retests of leakage rate have been conducted on the NS Savannah containment vessel.[38,52] The leakage rates indicated by these tests were consistent throughout a pressure range of 6 to 60 psig, with the leakage manifesting essentially laminar behavior at the low pressures and turbulent at the high pressures.

## 10.7 MAINTENANCE OF CONTAINMENT

The initial and periodic proof of containment integrity only partially fulfills the requirement that the containment system be effective whenever a potential radioactivity hazard exists. Careful design, construction, and inspection followed by these tests ensure, insofar as possible, that the containment is adequate, but they cannot guarantee that doors, valves, or other openings are not left open between tests, thus negating the efforts to provide effective containment. Therefore, in addition to performing proof tests, design safeguards and adequate procedures must be provided to give continuous assurance that containment integrity is being maintained when required.

### 10.7.1 Design Features

All containment designs provide certain features to ensure continuous containment integrity. Such design features may include interlocks, indicators, and controllers on all operating penetration accessories, such as the double doors of air locks, other access ports, and valves on piping that penetrates the containment barrier. All operating penetration accessories should be periodically tested to ensure their continuous reliable operation.

Interlocks are used to prevent one door of an air lock from being opened when the other door is open, to prevent a single equipment door or purge valve from being opened when the plant is operating or when the primary system is pressurized, or to prevent operation of other containment accessories that might invalidate containment integrity. On those types of containment that are normally inaccessible during operation and in which a small positive pressure is maintained, interlocks may be provided to prevent entry even through air locks when the plant is in operation or when the containment vessel is pressurized.

It may be possible to bypass an interlock in abnormal situations. In such cases, it is important that the plant operators know that the containment integrity is being affected. For these cases, indicator and alarm lights are normally provided in the control room. Routine checks of the correct operation of such indicators is necessary.

### 10.7.2 Operating Procedures

Besides having an adequately designed, constructed, and tested containment vessel and having all the necessary design features to ensure containment integrity, any plant must have strong administrative control through the mechanism of clearly written operating procedures. These should indicate to the operating personnel the importance of maintaining containment integrity and show how this is done. The operating personnel must, in turn, be thoroughly familiar with these procedures, since, in the final analysis, it is they who will be responsible for containment integrity. A typical procedure for maintaining containment integrity will specify the conditions under which access to the containment vessel is allowed, the responsible authority for approving and controlling such access, the conditions under which interlocks may be bypassed, the conditions under which single-door access may be used, and the positions of all valves and other penetration closures during both normal and abnormal operation. Routine checks should normally be provided to assure that all administrative control procedures are being properly followed.

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# 11. ECONOMICS

S. A. Bernsen* R. F. Griffin*

## 11.1 INTRODUCTION

The final selection of a containment design from among several alternate designs that meet the technical requirements for a particular plant and site is nearly always based on economic factors. In addition, economic aspects of containment and the cost of various containment features will become increasingly important as attempts are made (1) to improve containment effectiveness and reliability and thereby reduce the distance requirements for nuclear plant siting and (2) to compare the cost of varying degrees of containment effectiveness with the cost of transmitting power various distances from populated areas. The technical factors to be considered in designing and evaluating containment systems and various containment features are discussed in Chapters 8 and 9; this chapter discusses the economic factors. Its purpose is to promote an understanding of the aspects of containment which are of greatest economic significance and to provide guidance in making economic comparisons of alternate designs.

The limitations on presenting usable cost information must be emphasized at the outset. Accurate cost estimates can be made only on relatively well-developed designs for specific locations and with consideration of site conditions, labor costs at the construction site, and the general market conditions for equipment and materials. Thus, the costs of particular containment systems and the results of economic comparisons of alternate containment designs are valid only for the conditions for which they were developed. Great care must be taken when attempting to apply these costs and comparisons to different situations.

In view of these limitations, data on the cost of containment systems are presented in this chapter in three different ways: (1) various components of the containment system are identified and unit costs for these components are presented (sec. 11.3); (2) reported costs of containment systems which have been constructed are tabulated (sec. 11.4); and (3) conclusions presented in several independent comparative economic studies and evaluations of various types of containment systems are summarized (sec. 11.5). No attempt has been made to correlate the data from the various sources and no conclusions have been drawn concerning the economic advantages of particular types of plants or containment systems, since no generally applicable conclusions are apparent. The data presented reflect experience to date in reactor plant design and construction and, since more information is available for water-cooled reactor plants and for steel-shell containment vessels than for other types, this chapter contains more specific and complete data on these plants.

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*Bechtel Corporation, San Francisco, Calif.

In general, a containment system includes all components and systems in the plant that are provided to control, reduce, or eliminate leakage of fission products to the environment if they are released from the reactor in the event of a serious accident. The following items usually are included in the total cost of a containment system: the low-leakage concrete or steel enclosure around the reactor system, excavation and foundations for the containment structure, air locks or sealed doors and other special penetration closures necessary to maintain low leakage, interior finish and exterior insulation of the containment structure, and auxiliary systems such as a spray system (inside and outside) and the heating and ventilating system, including isolation valves, stack, and filters.

The Atomic Energy Commission has prepared a guide[1] for evaluating the cost of nuclear power. This guide establishes accounts for the various cost items in a nuclear plant. Wherever possible, tabulations and definitions of containment items used in this chapter correspond to the items in Account 219, Reactor Containment Structure, of the AEC Guide and to other appropriate account numbers, such as 212 G, Reactor Confinement Structure, and 221.5, Reactor Plant Containers in the Form of Tanks Installed Within a Building. Account 219 is reproduced here as Appendix F.

## 11.2 ECONOMIC IMPACT OF CONTAINMENT

### 11.2.1 General

The reactor containment system is one of a number of reactor plant systems provided to ensure that the plant can be operated without endangering the health and safety of the public. The cost of a typical containment system for a large plant, although significant, is not a major portion of the total plant cost, and it is unlikely that normal variations in containment requirements or containment design will have a major impact on overall plant economics. On the other hand, the requirement for containment affects the design of the entire plant and its location; thus the economic impact of containment is greater than that of the total cost of the items included in the definition of the containment system.

This section discusses the general economic impact of the requirement for containment, including effects that cannot be precisely evaluated but which are nonetheless real. Detailed information is not presented, but rather an attempt is made to place the subject of containment economics in proper perspective so that the cost information presented in the following sections can be used with suitable judgment.

### 11.2.2 Effect of Containment on Total Plant and Power Costs

It is of general interest to compare the total cost of the containment system with the total capital cost of the facility. In the case of power reactors, an evaluation of the cost of the containment system in terms of the cost of power is useful.

### 11.2.2.1 Capital Cost

Based upon published reports[2,3] and other sources of actual and estimated costs, an approximate range of total plant cost can be determined as a function of power. Figure 11.1 illustrates a typical relationship between total capital cost and power level for various types of reactor, principally water cooled. The costs indicated for the large plants now in planning stages are only approximations and, because of the incompleteness of published information, they are not necessarily comparable. Fig. 11.1 shows a broad cost range, which reflects the wide variation in plant design possible and the fact that many of the plants are of the developmental or demonstration type. As further experience is gained and more reactor plants are built strictly for economic power production, the range of variation in costs, as well as the total plant cost, should be reduced. The plants indicated by two points on Fig. 11.1 have increased their rated power since they were constructed, and therefore cost per kilowatt of electricity has decreased.

A range of containment costs is plotted as a function of power in Fig. 11.2 for various types of plant. Figure 11.2 is based on containment cost data from various sources, including references 2 and 4, the AEC cost evaluation guide,[5] and Table 11.10 in Section 11.4. The containment cost range is necessarily broad because of the wide variation in containment requirements and designs and because of the various ways in which containment costs are defined. Furthermore, a good correlation of containment cost with power level should not be expected, since containment vessel volume and design pressure are functions of coolant type, inventory, and stored energy more than they are of power level. An approximate evaluation indicated that for water-cooled reactors, the unit cost of containment is approximately 5w to 10w in dollars for kilowatt of electricity for spherical vessels and 10w to 20w for cylindrical vessels, where w is the pounds of coolant per kilowatt of electricity.[6] Obviously, such a relationship is not generally applicable, since cylindrical vessels have been economic in a number of cases, particularly in the lower power level systems.

By comparing Figs. 11.1 and 11.2, it can be seen that, for the plant size considered, containment may account for as little as 5 to more than 20% of the total plant cost, the percentage generally being smaller the larger the plant.

### 11.2.2.2 Power Cost

The effect of containment on power cost will vary with the nature of the utility but can be defined with two variables: capital charge rate and plant load factor. Normally, capital charge rates range from 5.5 to 8.0% for publicly owned utilities and from 11 to 16% for privately owned utilities. Plant load factors range from 50 to 90%.

In order to establish a typical set of factors to illustrate the significance of containment cost, it is assumed that the total cost of concainment is $\$4 \times 10^6$ for a 200-Mw(e) power plant of \$20/kw(e). Also, it is assumed that the plant is operated at an 80% load factor by a privately owned utility, and the yearly cost for capital investment is 14%.

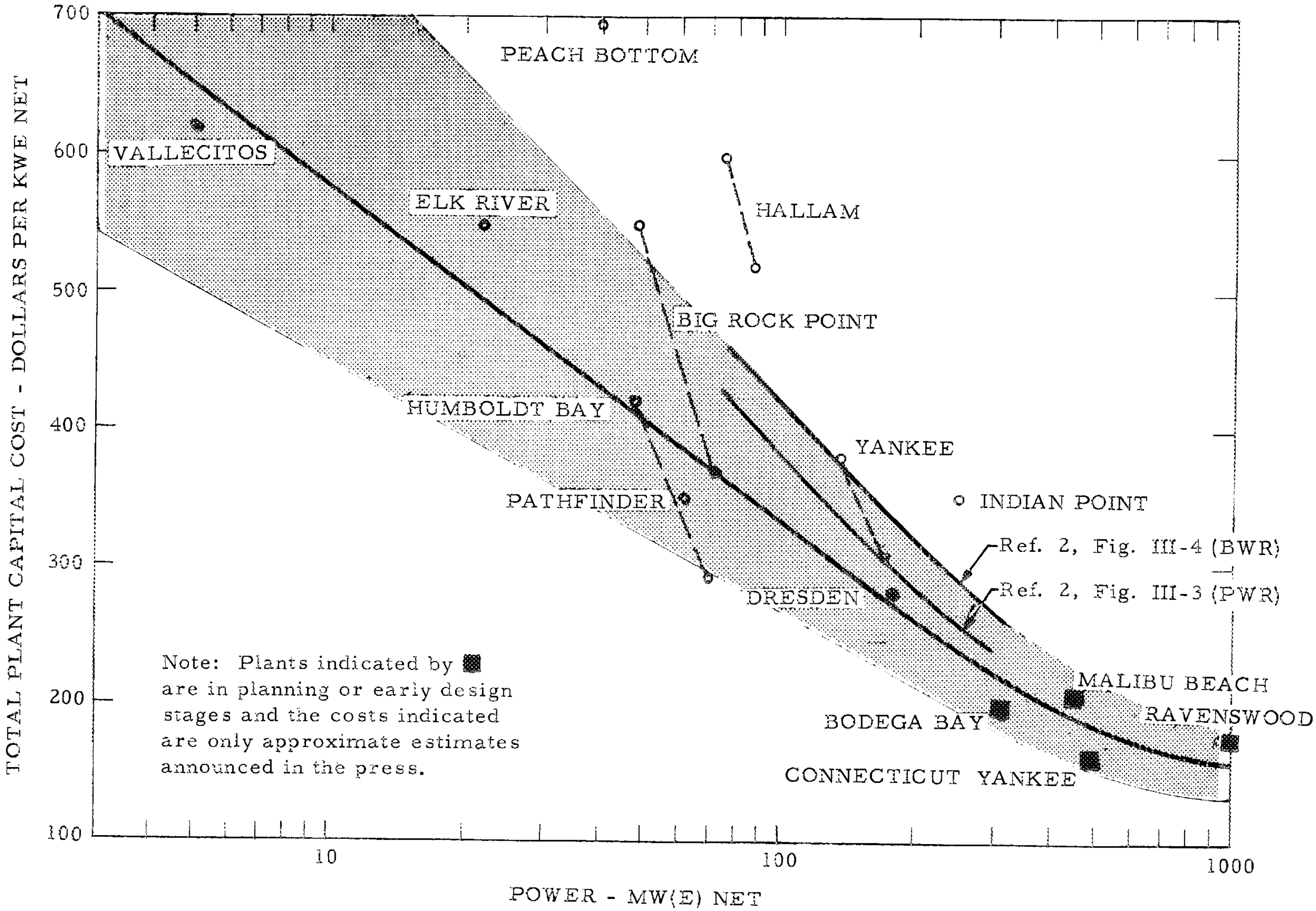


Fig. 11.1. Total Plant Capital Cost vs Power.

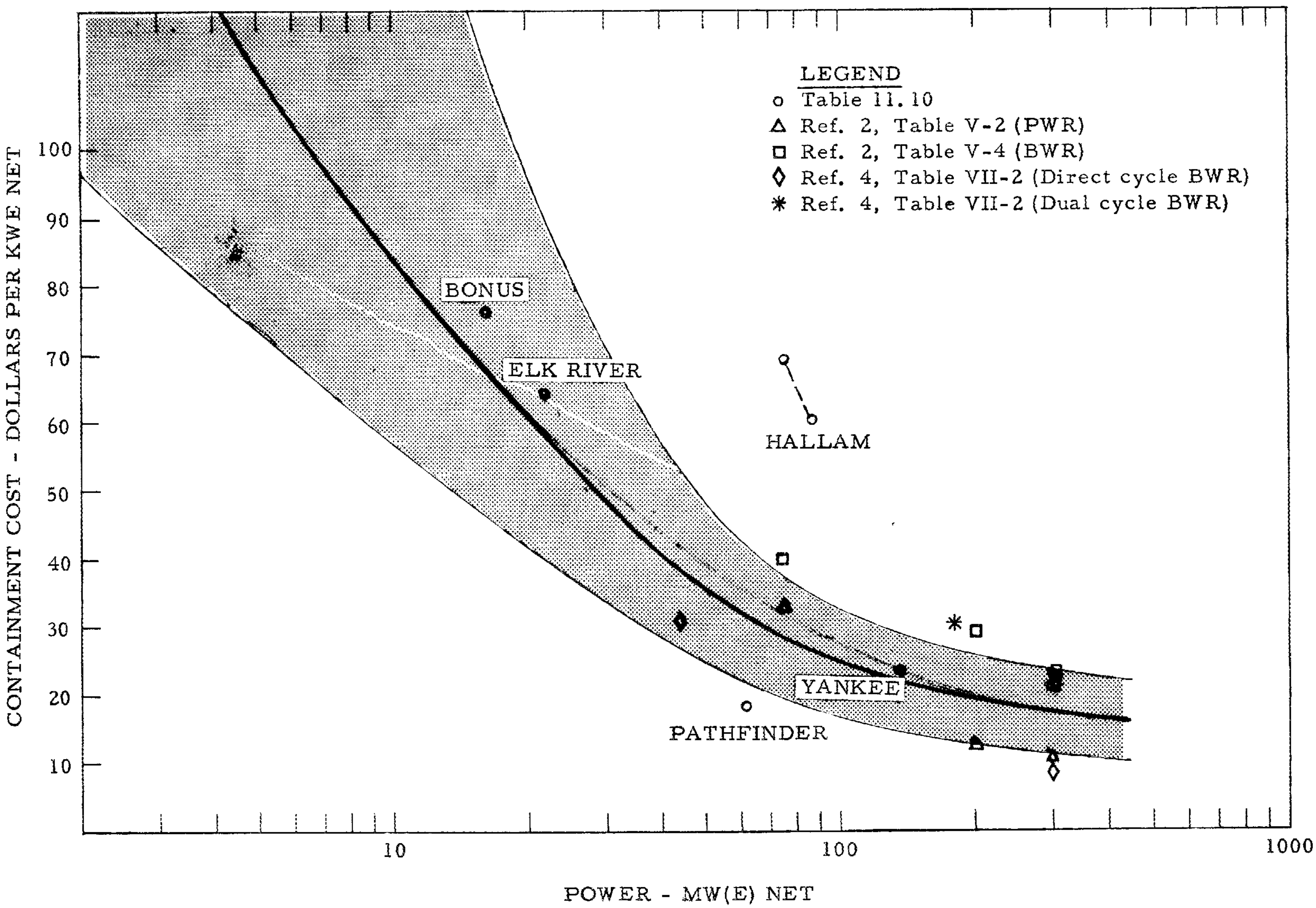


Fig. 11.2. Containment Cost vs Reactor Power.

The containment requirements result in a contribution to the power cost of:

$$\frac{\$20 \times 14\%}{8760 \times 80\%} = \$0.40 \times 10^{-3} \text{ or } 0.40 \text{ mills/kwhr}$$

Since the total cost of power from such a plant may be in the range of 6 to 8 mills/kwhr, the containment system could account for more than 5% of the cost of power.

#### 11.2.2.3 Containment Cost Penalty

The previous example employs costs that reflect the total cost of a containment system. These include the entire reactor enclosure and other structural components that would be required to house the plant whether containment was required or not. Thus, it results in a cost penalty that is larger than can be attributed to the requirements for containment alone. It is necessary to subtract from the total containment cost the value of the conventional structural components the containment replaces in order to determine the true containment cost penalty. In the above example, the conventional counterpart to the containment structure might cost as much as $\$2 \times 10^6$ or about half of the cost for the containment system; hence, the containment cost penalty, or the additional cost due to the requirement for containment, would be about 3% of the total power cost. These typical values illustrate that variations in the cost of containment over normal ranges cause much less significant variations in the overall plant economics.

### 11.2.3 Effect of Containment on Plant Location

The effectiveness and reliability of the containment system and the costs associated with it are important considerations in determining the location of a nuclear plant. Although distance from populated areas has been the principal consideration in siting nuclear facilities thus far, it is apparent that economical large-scale production of nuclear power will require large nuclear plants to be built reasonably close to population centers. Therefore, it is to be expected that increasing emphasis will be placed on the additional containment features required to permit use of less remote sites. The cost of providing this increased containment effectiveness can then be balanced against the savings in cost of power transmission to provide some guidance for determining optimum plant location.

#### 11.2.3.1 Engineered Safeguards vs Distance Requirements

The AEC Reactor Site Criteria[7] and report on "Calculation of Distance Factors for Power and Test Reactor Sites"[8] provide guidance for determining suitable distances of nuclear plants from populated areas. Reference

8 gives examples of required distance as a function of power level for typical reactor plants with certain assumed containment features. These guides acknowledge that additional "consequences-limiting" engineered safeguards can compensate for undesirable site characteristics and can reduce the distance requirements. Beck[9] discussed the requirements for such engineered safeguards. He stated, in effect, that since it cannot be guaranteed that a fission-product releasing accident within the containment system cannot occur, and since there must be certainty that no more than a very small fraction of the fission-product inventory will escape to the environment, these safeguards must have a very high degree of dependability. In fact, the confidence in the engineered safeguards must be comparable to that in the distance factors they replace.

While it may be that none of the containment systems provided for existing plants would meet the requirements for locating a large plant in a heavily populated area, there is no reason to believe that such a system could not be built. In evaluating the Malibu Beach site proposed for the Los Angeles Department of Water and Power nuclear power plant, the AEC's Advisory Committee on Reactor Safeguards concluded that, in spite of the "stringent requirements imposed by the site," either a pressurized-water reactor or a boiling-water reactor "if provided with adequate containment of the primary system can be located at the site with reasonable assurance that the reactor can be operated without undue risk to the health and safety of the public."[10] Similarly, for the Haddam-Neck site proposed for the Connecticut Yankee plant, the ACRS stated: "Because the Haddam-Neck site does not meet the present site distance criteria, reliance must be placed on proved engineered safeguards. In the case of this site, the ACRS believes the added control needed for protection of the health and safety of the public can be accomplished by the applicant."[11]

Johnson[12] describes three "consequence limiting" containment features that should be suitable for large nuclear plants located in or near metropolitan areas. These are pressure suppression, underground construction, and multiple-barrier containment. Containment shielding will probably be required for plants in metropolitan areas but it may be required at some more remote locations as well, depending upon the existence of other power facilities at the same site and on the presence of normally occupied areas adjacent to the site.

The extent of reduction that might be realized in the required distances from populated areas as a result of additional engineered safeguards was discussed by Ergen,[13] particularly with reference to multiple-barrier containment. Ergen uses the concept of a "safeguard factor," the factor by which the power of a given reactor can be increased at a given site (as calculated in ref. 8), by the addition of engineered safeguards. Since the distance requirements are related to the 2/3 power of reactor power, this is also equivalent to a reduction in the required distances for the same power. As an example, using as a basis the examples and assumptions in ref. 8 and assuming containment shielding from direct radiation, Ergen shows that reducing the iodine hazard so that it is equivalent to the hazard from noble gases will result in a safeguard factor of 23 if the exclusion radius is limiting and a factor of 153 if the outer boundary of the low population zone is limiting. Reducing the iodine release still further and holding up the noble gases to allow for some decay will further increase the safeguard factor. Of course, a valid safeguard factor depends on many things, including proper evaluation of the dependability and actual effectiveness of the safeguard features, as well as detailed consideration of the behavior of other fission products that will become relatively more important as the hazards of the normally limiting ones are reduced. It may be possible, however, to evaluate by similar methods the relative merits of various containment features so that an approximate relationship between containment effectiveness and cost can be determined.

### 11.2.3.2 Transmission Costs

Power transmission costs, against which the cost of greater containment effectiveness must be weighed, depend on many factors and vary over a wide range. Both the installed cost of the lines, structures, and terminal equipment and the operating costs, including line losses, must be considered, since both are economic penalties of a remote plant location.

Installed costs of transmission lines include the cost of the substations and transformers and other associated equipment at each end of the line. These usually are a substantial portion of the total line cost, which decreases with increasing length of line. However, in estimating the cost penalty associated with greater transmission distance, it is the incremental cost of transmission that is most significant, and the relatively constant cost of the terminal equipment can be neglected for comparison purposes. Right-of-way costs are a significant part of the incremental costs and can vary over a very wide range, particularly when considering both rural and metropolitan areas. Underground lines have, of course, much higher equipment and installation costs. A range of installed cost per mile of extra-high-voltage transmission lines of various voltages and types is presented in Table 11.1. These figures were taken from a report by the Federal Power Commission National Power Survey[14] and are typical costs at 1962 levels based on averaging costs from all over the country. Since most transmission lines traverse rural areas, it can be assumed that the right-of-way costs are representative of such areas.

Operating costs of power transmission can vary even more widely than the installed costs of the lines, depending not only upon the type of equipment but also upon the characteristics of the utility's transmission and distribution system, load factor, financing arrangements, etc. Table 11.2 presents extra-high-voltage transmission costs in mills per kwhr received, including line losses and investment costs on the lines and facilities, for representative systems at various voltages and loads for distances of 100 and 200 miles and for load factors of 50 and 85%. These costs were also taken from a Federal Power Commission National Power Survey report[15] and represent the cost of point-to-point transmission of power by a single circuit, with no provision to ensure firm transmission conditions. They are thus not necessarily representative of actual power system operation.

In a study for the 255-Mw(e) Indian Point plant, Kallman and Hanson[16] indicated incremental transmission line costs of $2,000,000 for 30 miles or $67,000 per mile in a rural area. In announcing the 1000-Mw(e) Ravenswood plant, at one time proposed for metropolitan New York, Consolidated

Table 11.1. Installed Costs of Extra-High-Voltage Power Transmission Lines

| Type | Voltage (kv) | Structure | Cost per Mile: Right of Way and Clearing | Cost per Mile: Labor and Material | Cost per Mile: Total Cost |
|---|---|---|---|---|---|
| Overhead, ac | 230 | Wood | $10,000 | $ 35,000 | $ 45,000 |
| | 230 | Steel | 10,000 | 45,000 | 55,000 |
| | 230 | Steel | 10,000 | 60,000 | 70,000 |
| | 345 | Wood | 12,000 | 48,000 | 60,000 |
| | 345 | Steel | 12,000 | 65,000 | 77,000 |
| | 345 | Steel | 12,000 | 86,000 | 98,000 |
| | 500 | Steel | 14,000 | 85,000 | 99,000 |
| | 700 | Steel | 18,000 | 125,000 | 143,000 |
| Overhead, dc | ±250 | Steel | 10,000 | 56,000 | 66,000 |
| | ±375 | Steel | 12,000 | 68,000 | 80,000 |
| | ±500 | Steel | 14,000 | 78,000 | 92,000 |
| Underground, ac | 138 | | | 327,000 | 327,000 |
| | 230 | | | 359,000 | 359,000 |
| | 345 | | | 697,000 | 697,000 |
| | 500 | | | 1,056,000 | 1,056,000 |
| Underground, dc | ±250 | | | 264,000 | 264,000 |
| | ±375 | | | 634,000 | 634,000 |

Edison indicated that to build the plant out of town would require an additional cost of $75,000,000 for transmission lines,[17] most of which would have to be underground.

In a study aimed specifically at determining the extent to which reactor site selection criteria impose a power transmission cost differential on nuclear power plants, 35 utilities in high fuel cost areas were asked to estimate the transmission cost penalty if proposed conventional power plants had to be relocated to meet the population distance criteria for nuclear plants.[18] As would be expected, the replies varied over wide ranges and no general correlations or meaningful averages were possible. The estimated costs for the transmission lines and terminal equipment varied from $12,000 per mile to $260,000 per mile. Right-of-way estimates varied from $2,000 per mile to $400,000 per mile. The average investment cost penalty for transmission lines to relocate the plant an average distance of 30 miles farther from the load center was about $2,500,000. Assuming line losses at 8% per 100 miles, a 300-Mw load, a fixed charge rate of 14%, and an 80% load factor results in an additional penalty for losses equivalent to an investment of about $2,000,000.

Table 11.2. Extra-High-Voltage Power Transmission Costs as Affected by Distance and Load Factor

| Voltage (kv) | Type | Load (Mw) | Transmission Cost (mills/kwhr received) | | | |
|---|---|---|---|---|---|---|
| | | | 100 Miles | | 200 Miles | |
| | | | 50%[a] | 85%[a] | 50%[a] | 85%[a] |
| 345 | ac | 250 | 1.1 | 0.65 | 1.9 | 1.15 |
| 345 | ac | 500 | 0.65 | 0.4 | 1.25 | 0.8 |
| 500 | ac | 250 | 1.3 | 0.65 | 2.35 | 1.3 |
| 500 | ac | 500 | 0.7 | 0.4 | 1.25 | 0.75 |
| 500 | ac | 1000 | 0.45 | 0.25 | 0.75 | 0.5 |
| 700 | ac | 500 | 0.9 | 0.5 | 1.8 | 1.1 |
| 700 | ac | 1000 | 0.5 | 0.3 | 1.0 | 0.6 |
| 700 | ac | 2000 | 0.4 | 0.25 | 0.65 | 0.4 |
| ±250 | dc | 600 | 2.05 | 1.3 | 2.4 | 1.55 |
| ±375 | dc | 900 | 2.1 | 1.3 | 2.35 | 1.45 |
| ±500 | dc | 1200 | 2.15 | 1.35 | 2.4 | 1.5 |

[a]Load factor.

It appears that several million dollars in transmission costs could be saved by locating the plant several miles closer to a load center in a metropolitan area. Some part of this saving could be applied to containment costs if such a relocation was warranted on other bases.

### 11.2.3.3 Site Costs

The cost of site acquisition must also be considered in comparative evaluations. The cost of the site may vary by several orders of magnitude and is markedly dependent on local conditions. A site in a populated area will probably cost more than a remote site, so the additional cost of power transmission may be partially compensated for. However, differences in site costs usually are small compared with differences in transmission costs. Site costs also may be somewhat dependent on the degree of containment shielding provided. If direct radiation at the site boundary is limiting, either containment shielding or a large site may be necessary.

### 11.2.3.4 Evaluation

A thorough economic and technical evaluation of each situation is necessary to determine the extent to which more favorable siting can justify additional containment expense. More precedent than now exists is required before a relation between containment cost and containment effectiveness can be established for a given case. However, it seems clear that containment costs much greater than those indicated on Fig. 11.2 could be justified if plants thereby could be located closer to metropolitan areas.

### 11.2.4 Effect of Containment on Plant Design and Construction

In addition to the costs directly associated with the containment system components, other plant costs are incurred indirectly because of the requirement for containment. These indirect costs cannot be readily identified or evaluated, but they may be quite significant, and it is important to know that they exist. It is largely because of these less obvious costs that a meaningful economic comparison of alternate containment designs requires relatively complete estimates of the entire plant, rather than simply a comparison of the costs of identifiable containment components.

#### 11.2.4.1 Design Changes

The use of containment will normally impose restrictions on the location of equipment both within and outside the containment structure. In most cases, it is not possible to employ the most economical physical arrangement of all plant components; thus, some economic penalty in plant design will result directly from containment requirements. For example, in cases where the turbine-generator is included within the containment enclosure, the conventional arrangement of the turbine-generator and the condenser may not be possible. In addition, special provisions are required to bring condenser cooling water and the generator power bus through the containment shell. The power bus, as well as other electrical and instrumentation leads, are likely to be somewhat longer than are normally required in a conventional power plant. Depending upon the degree of access to the containment permitted, increased use of automatic and remote equipment may be required. In cases where the turbine generator equipment is outside the containment structure, some but not all the special features described above can be eliminated. On the other hand, the main steam lines may be substantially lengthened and, due to increased pressure drop, may be increased in diameter. Additional stop valves may be required in the lines that penetrate the containment enclosure to isolate the portion of the plant within the containment structure from that portion external to it. In direct-cycle plants where the steam line is part of the primary system, fast-operating double-isolation valves are required to ensure rapid isolation of the plant in the event of an accident.

Reactor plant primary system components are normally arranged compactly near the center of the containment structure to minimize the amount of primary shielding required. However, in an underground or shielded containment design it may be possible to utilize the containment shielding as primary shielding and permit a more open plant arrangement without increasing the shielding material required. Such variations in plant arrangement could affect future operating and maintenance costs, as well as equipment and construction costs.

Plant arrangement may also be affected by the requirements for refueling. The height of the entire plant with respect to grade may be governed by the need to move a refueling cask into the containment vessel or refueling building.

It is apparent that the effect of these variations in design cannot be evaluated through an assessment of costs of individual, identifiable containment components. Thus, although the effects of changes in design of the rest of the plant should be recognized by the evaluator when comparing relative costs of alternate containment concepts, it is difficult to assign specific values to these effects.

#### 11.2.4.2 Construction Requirements

In addition to the design changes incurred as a result of containment, containment requirements will also influence methods, procedures, and schedules employed in construction of the plant. This can affect the handling equipment required and the use of the labor force. For example, in single-stage construction (see Chap. 8), pressure testing of the containment shell is required before sections of the shell are made inaccessible by concrete pours. In this case, all internal concrete work and the installation of the major equipment, including the reactor vessel, would have to be accomplished through a temporary opening in the containment shell. This requirement may have a substantial effect on the cost of placing concrete and on the cost of special equipmemt for handling the heavy components within the containment shell. On the other hand, multiple stage construction, sometimes used to avoid some of these problems, raises some additional problems. For example, erection of the steel shell is interrupted and piping and electrical connections must be delayed.

The containment design may also produce highly confined areas in which work is very difficult and in which special ventilation systems and other temporary facilities may be required. Such congested conditions limit use of the labor force and thus increase construction costs.

## 11.3 UNIT COSTS OF CONTAINMENT COMPONENTS

### 11.3.1 General

Unit costs for various containment components, including those which account for most of the cost of a containment system, as well as those of less economic significance, are presented in this section. Where possible, these costs are based on actual construction cost experience or on detailed cost estimates of particular containment designs. In general, they represent total installed cost, including material and equipment costs, labor, and other field costs. Since all costs will vary from one application to another, cost ranges are presented for many of the items to indicate the expected range of variation. Where single costs are given, these should be considered to be typical or average values only.

The principal value of presenting these costs is to indicate the relative costs of various containment system components and thereby to promote an understanding of the aspects of containment that are of greatest economic significance. It may be possible to use these unit costs in developing a rough estimate of the total cost of a particular containment system or in making an economic comparison of alternate containment systems; however, if this is done, it should be realized that even if the costs presented in this section are considered sufficiently accurate for these purposes, the list of items is not necessarily complete, and all costs are not necessarily additive. An estimate or an economic comparison based on these costs should be used with care and with a full understanding of its limitations.

### 11.3.2 Steel Containment Vessels

The containment enclosure itself is normally the most significant single cost component in a containment system. The enclosures used in most of the containment systems built in the United States to date have been large steel vessels of various shapes, principally spheres or cylinders. This section presents approximate unit costs of various types of steel containment vessels based on the actual costs of several such vessels. Some generalized curves, based on average steel costs and on the approximate wall thicknesses required at various pressures for cylindrical and spherical steel containment vessels of various sizes, are presented in Section 11.5.

A steel containment vessel is usually furnished by a steel fabricator as a complete package, including all materials, fabrication, erection, and testing. The vessel, as supplied, usually includes air locks but has the other penetrations blanked off. Based on cost data for several steel containment vessels ranging in volume from 1,100,000 to 3,600,000 $ft^3$, Fig. 11.3 was prepared to show a reasonably well-correlated curve of cost per cubic foot of contained volume versus design pressure for steel vessels in this size range. Note that at lower design pressures the cost per cubic foot approaches a constant value as external forces, rather than internal pressure, begin to determine minimum plate thickness. The costs plotted in Fig. 11.3 correspond to those usually included in Account Nos. 219.42 and 219.43 of the AEC Cost Evaluation Guide.

The costs per cubic foot of contained volume shown in Fig. 11.3 correspond to unit costs of approximately 0.45 to 0.60 dollars per pound of steel in the completed vessel. This cost per pound is highest for low-pressure, low-volume vessels and lowest for the larger vessels and those with higher design pressures.

As an indication of the total costs associated with steel containment vessels, the total cost of all items included in Account No. 219 plus all internal biological shielding concrete is plotted as a function of design pressure in Fig. 11.4 for vessels of the same volume range as in Fig. 11.3.

Steel pressure vessels are also used in pressure-suppression containment systems, although the configurations and design conditions are considerably different from those of conventional steel shell containment vessels. Because there has been less actual experience with this type of containment to date, little information is available on the cost of the steel components for these systems. For a 60-Mw(e) natural-circulation

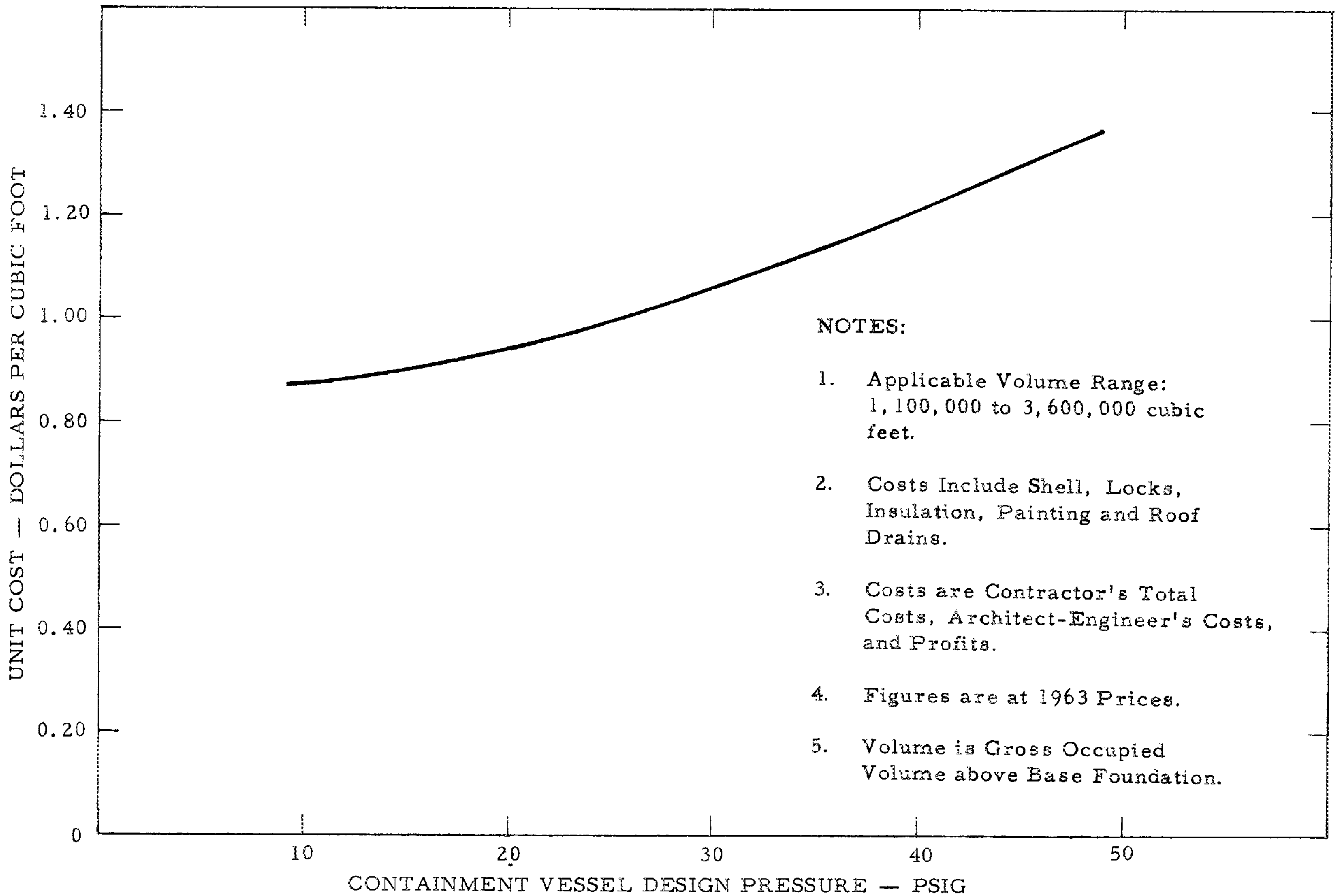


Fig. 11.3. Steel Containment Vessel Unit Cost as a Function of Design Pressure.

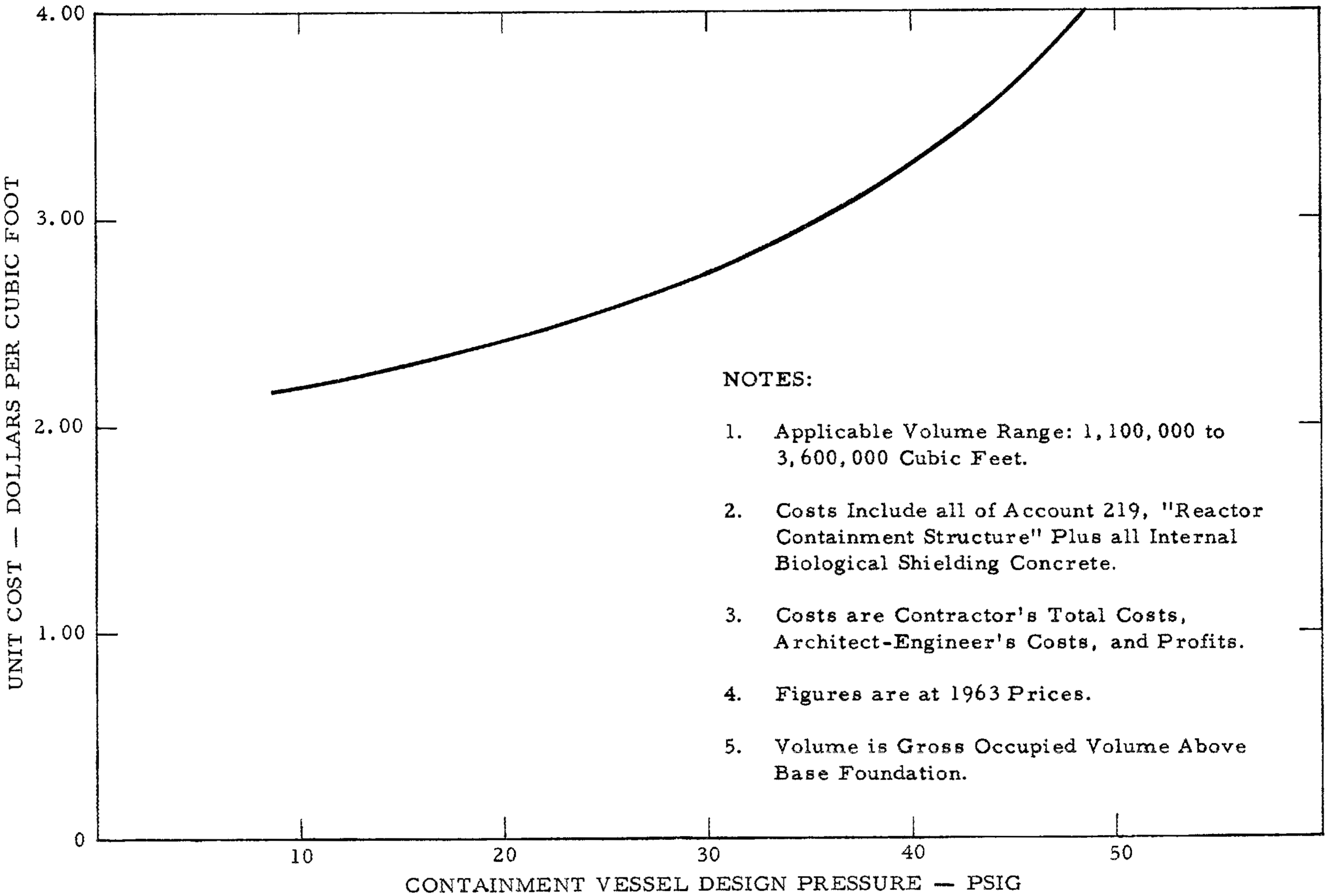


Fig. 11.4. Total Containment System Unit Cost as a Function of Vessel Design Pressure.

boiling-water reactor the cost of the steel containment components was estimated to be \$300,000. For a 400-Mw(e) boiling-water reactor the cost of the steel drywell, pressure-suppression piping, and torus-shaped suppression pool was estimated at \$1,500,000.

### 11.3.3 Concrete Containment Vessels

Few concrete containment vessels have been built, but interest in containment structures of this type is increasing. Figure 11.5 shows the approximate cost per cubic foot for concrete containment vessels of two different sizes. The cost includes a steel-plate liner, which is likely to be required for all concrete containment vessels to provide the necessary degree of leak tightness. In the designs used as a basis for Fig. 11.5, the steel liner was also used as the internal form for the concrete. The costs shown in Fig. 11.5 also include the cost of the foundation for the site conditions indicated. For a concrete vessel, the foundation is part of the vessel and usually is not estimated separately. The concrete walls were designed for minimum allowable structural thickness (1 1/2 to 2 ft), with no additional concrete added for containment shielding. An important advantage of concrete containment vessels is that concrete for containment shielding can be added at a relatively low incremental cost (see sec. 11.3.4).

The costs shown on Fig. 11.5 apply to the specific conditions assumed as the basis for the design and may vary considerably for other assumed conditions. Because the cost of the foundation is included, these costs are sensitive to variations in site soil conditions, as are costs of all types of foundations. Also, regional variations in wage rates have a greater effect on the cost of concrete containment vessels than they do on the cost of steel containment vessels because of the greater amount of manual labor required at the site.

Concrete structures can also be used for caisson-type containment systems at sites with granular soil, a high ground-water table, and no underground obstructions. The estimated costs of caissons will vary widely, depending upon the site conditions and the uncertainties associated with sinking a caisson. A containment structure using a caisson with a displaced volume of 11,000 $yd^3$ and a free volume of 5,000 $yd^3$ (135,000 $ft^3$) was estimated to cost \$1,200,000, including the cost of sinking, or approximately \$100 per cubic yard displaced and \$240 per cubic yard (\$9 per cubic foot) of free volume. This structure was designed for an internal test pressure of 20 psig, but for a caisson the structural design is determined primarily by the weight required for sinking and by the bending moments that could occur during sinking rather than by the internal pressure. For a larger plant, a caisson with a displaced volume of 100,000 $yd^3$ and a free volume of 40,000 $yd^3$ was estimated to cost \$5,000,000, or \$50 per cubic yard displaced and \$125 per cubic yard (less than \$5 per cubic foot) of free volume.

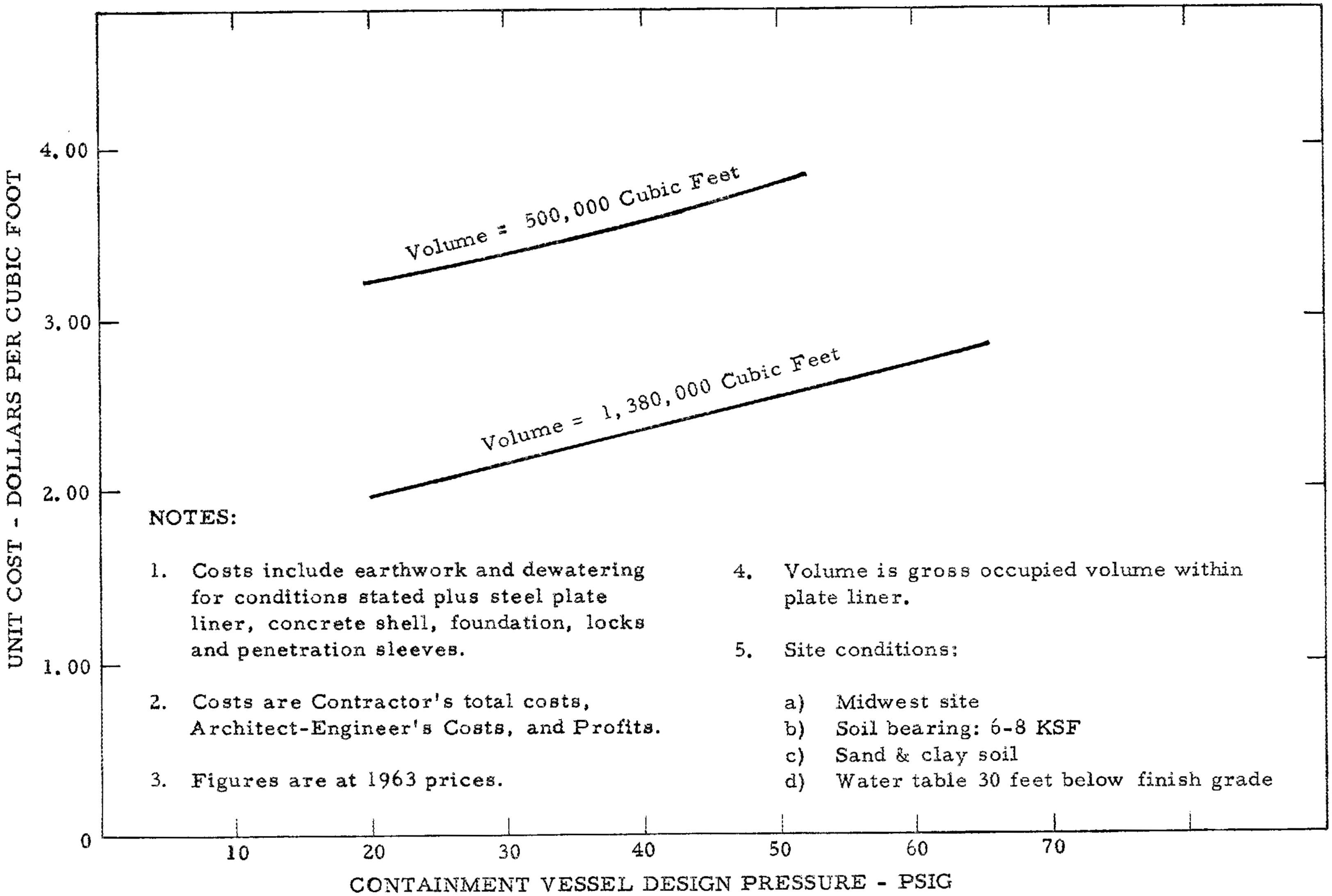


Fig. 11.5. Concrete Containment Vessel Unit Cost as a Function of Design Pressure.

### 11.3.4 Other Concrete Structures

Construction with concrete involves a series of independent operations, including erecting forms, placing reinforcing steel, placing and finishing the concrete, and removing and cleaning the forms. Since the concrete work in nuclear plants varies from simple forms to intricate structures with many penetrations, and includes concrete both within and external to the containment shell, a wide range of concrete costs is to be expected. The cost ranges presented in Table 11.3 for different applications of concrete may be considered representative for concrete in and around steel-shell containment vessels constructed in a single stage. The cost of concrete placed inside the shell may be less where multistage construction is used. The unit costs of concrete for similar applications in other than steel shell types of containment structures may fall outside these ranges, depending upon the amount of concrete of each type used.

Table 11.3. Cost Ranges for Emplaced Concrete

| Application | Unit Cost ($\$/yd^3$) |
|---|---|
| Structural concrete external to the shell | 40–100 |
| Lean or nonstructural concrete external to the shell | 20–30 |
| Heavy concrete foundation inside the shell | 60–75 |
| Other concrete work inside the shell | 100 and up |

The incremental cost of adding concrete for shielding to a concrete containment vessel may be in the range of $20 to $35 per cubic yard. This is less than the unit cost of the materials in the basic concrete vessel, since little additional reinforcing or form work is required. However, the foundation cost will also increase when concrete is added for shielding, so the total increase in cost will be greater than just the incremental cost of the concrete for the wall. The foundation cost increases will be strongly dependent on site conditions.

### 11.3.5 Excavation

The cost of excavation may be a significant cost item, particularly for underground containment designs or for containment structures extending well below grade. Excavation costs, even more than other construction costs, are very sensitive to site conditions and a wide range of unit costs is to be expected. The cost of large, open excavations varies from about 30¢ per cubic yard in alluvial soil and sand to $10 per cubic yard where rock is encountered.

### 11.3.6 Air Locks and Doors

The air locks and doors used for personnel and equipment access into containment vessels are significant cost items, but in many cases they are included in the cost of the containment vessel package discussed in Section 11.3.2. The costs of air locks and doors vary widely for different sizes, shapes, design pressures, operating mechanisms, and other design details; however, the cost ranges given in Table 11.4 are typical for the types of air locks and doors used most frequently in containment vessels for which access during plant operation is required.

Table 11.4. Typical Costs of Air Locks and Doors

| Type | Size (ft) | Unit Cost ($) |
|---|---|---|
| Autoclave door | 7 1/2 | 14,000–18,000 |
| Autoclave door | 10 | 22,000–26,000 |
| Escape lock | 2 1/2 | 20,000–30,000 |
| Personnel access lock, power operated | 7 | 40,000–70,000 |
| Equipment access lock, power operated | 10 | 80,000–100,000 |

### 11.3.7 Heating, Ventilating, and Air Conditioning

The heating, ventilating, and air conditioning system may represent a significant portion of the containment system cost. The cost is largely a function of the volume and heat loads within the containment vessel under abnormal as well as normal conditions but will vary with plant size and type and with local climatic conditions. The cost per cubic foot of containment volume, based on actual and estimated costs of several representative containment systems, is plotted in Fig. 11.6. The costs are greater than for comparable systems in conventional power plants because of the requirements for isolation valving, radiation monitoring, higher pressure ductwork, special filtering equipment, and in many cases installation of the system in more congested areas.

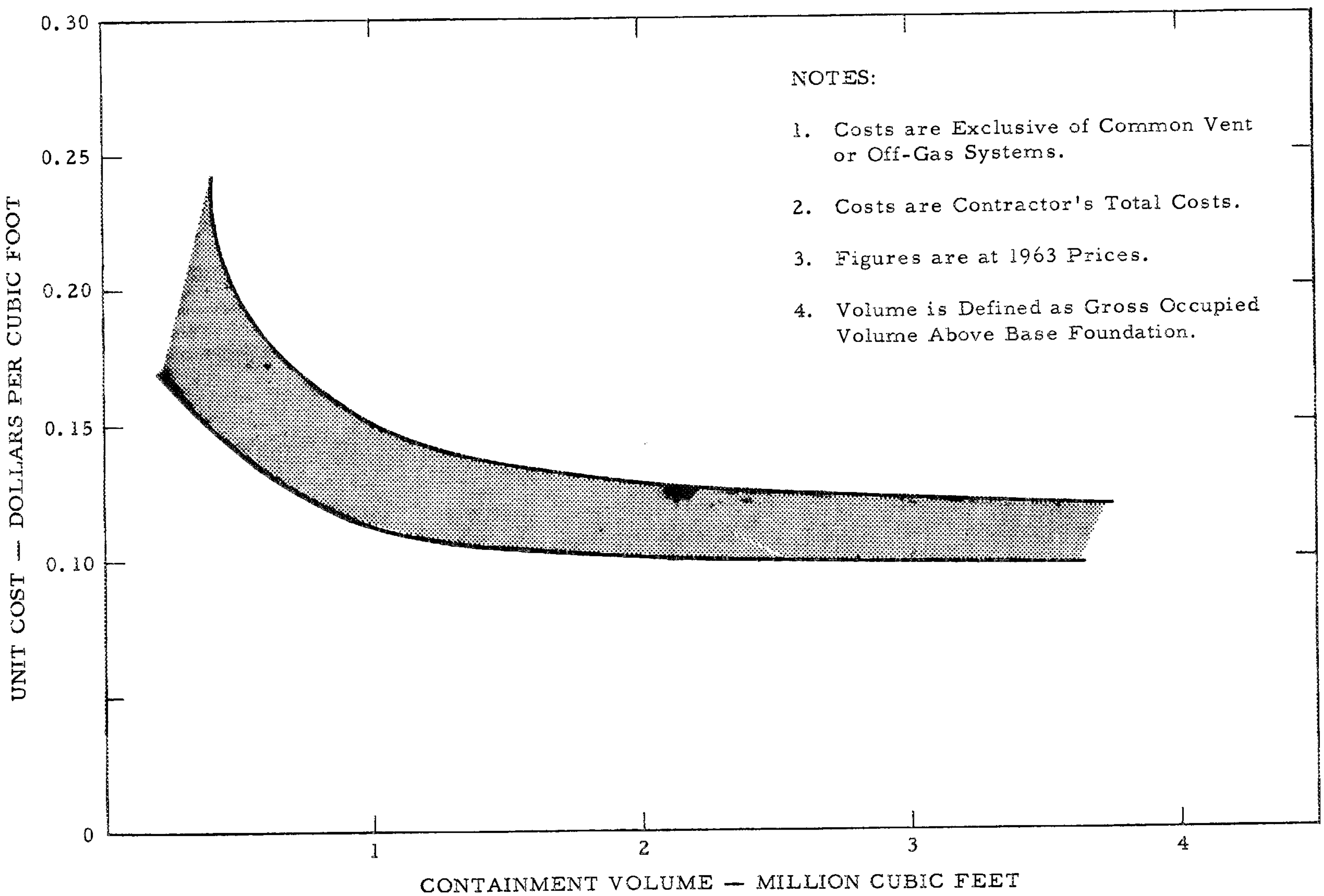


Fig. 11.6. Containment Heating, Ventilating, and Air Conditioning System Unit Cost as a Function of Containment Volume.

### 11.3.8 Filters

Filters and other fission-product removal or trapping systems are included in many containment ventilation systems. In addition, recirculation filters may be provided to remove fission products from the containment vessel atmosphere in the event of a nuclear accident. Usually the basic filter elements are a minor portion of the overall ventilation system cost, the major portion of the cost being associated with the ductwork, instrumentation and controls, and blowers. Although there are many types of filters and other fission-product removal devices available and their costs can vary widely, the types listed in Table 11.5 can be considered typical.

Table 11.5. Filter Costs[a]

| Fission-Product Removal Device | Capacity (cfm) | Unit Cost ($) |
|---|---|---|
| Roughing filter | 1,000 | 10 |
| Activated-charcoal filter | 1,200 | 1,600 |
| High-efficiency filter | 1,200 | 100 |
| Wet-caustic scrubbing tower | 2,000 | 800-1,600 |
| Wet-caustic scrubbing tower | 10,000 | 2,200-4,500 |
| Wet-caustic scrubbing tower | 50,000 | 8,000-21,000 |

[a]Excluding installation costs.

The activated charcoal filter listed in Table 11.5 is an extended-surface type of filter and is usually used in conjunction with the high-efficiency filter also listed. The wide price ranges for the caustic scrubbing towers reflect variations in designs depending upon the efficiency desired and the characteristics of the contaminated air. Approximate total annual costs of complete filter systems, including maintenance and filter replacement costs, are presented in Section 11.5.9.

### 11.3.9 Stacks

The stacks provided on most nuclear plants are part of the containment ventilating system but they are well-defined items and therefore are listed separately. The variation of the cost of a stack as a function of stack height is plotted in Fig. 11.7 for concrete stacks typical of those used for most nuclear plants. The cost of the stack foundation is not included.

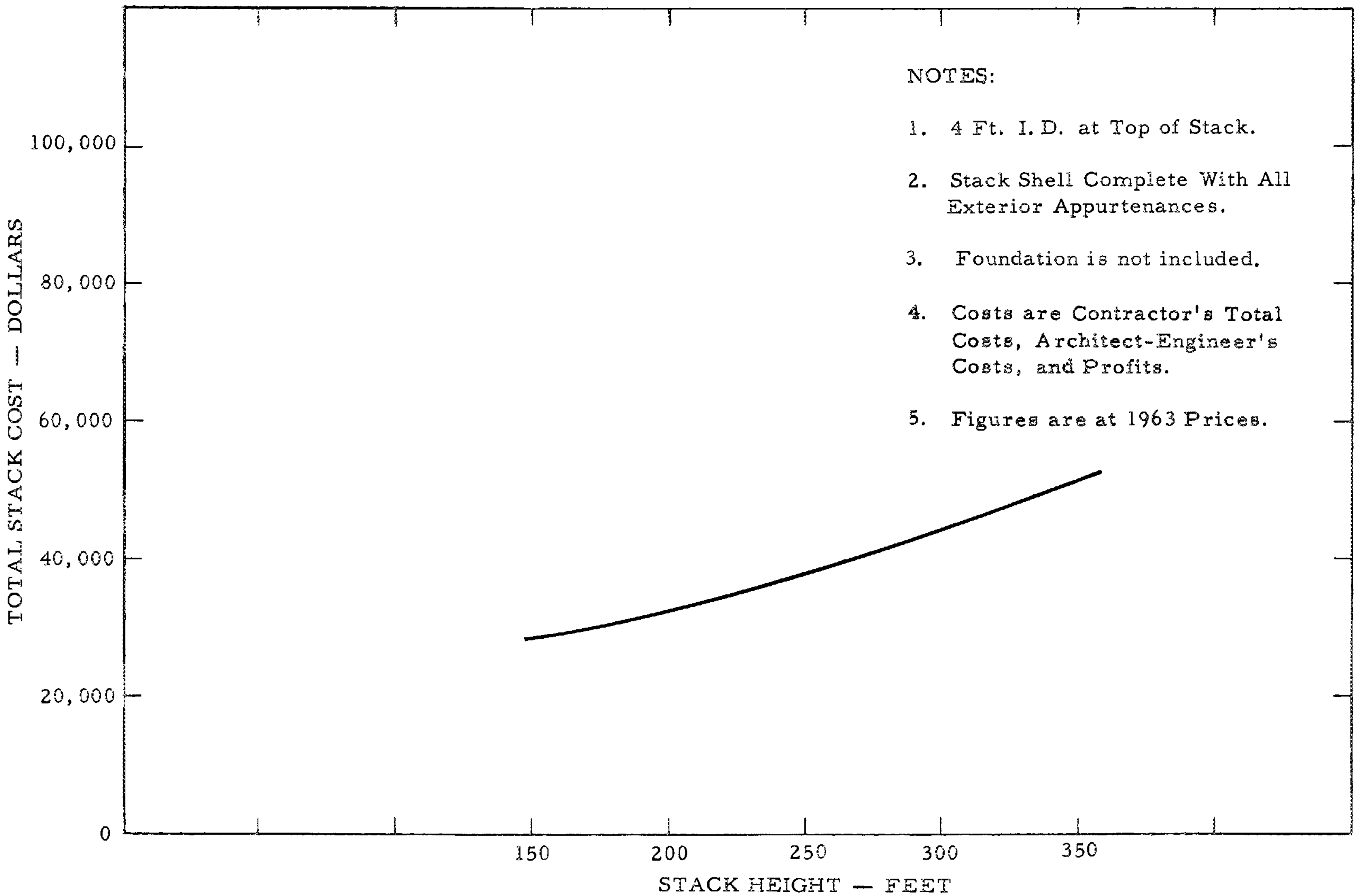


Fig. 11.7. Cost of Concrete Stack as a Function of Stack Height.

## 11.3.10 Penetrations

The cost of electrical and piping penetrations of containment vessels may add up to a significant total because of the large number required. A wide variety of penetration designs have been used, some of which are described in Chapter 9. Piping penetrations for steel containment vessels generally consist of two parts: the nozzle, which with its reinforcing is an integral part of the steel shell, and the attachment of the pipe to the nozzle.

The nozzles are usually supplied as part of the containment vessel package and are provided with temporary end caps to blank them off during testing of the vessel. The cost added to the basic containment vessel cost for a typical set of penetration nozzles through a 1-in.-thick containment vessel wall is given in Table 11.6.

Table 11.6. Costs of Penetration Nozzles

| Nozzle Diameter (in.) | Number Required | Unit Cost ($) | Total Cost ($) |
|---|---|---|---|
| 30 | 2 | 1500 | 3,000 |
| 26 | 2 | 900 | 1,800 |
| 24 | 20 | 800 | 16,000 |
| 22 | 100 | 600 | 60,000 |
| 20 | 100 | 550 | 55,000 |
| 12 | 2 | 350 | 700 |
| 10 | 4 | 300 | 1,200 |
| 8 | 8 | 250 | 2,000 |
| 6 | | | |
| 4 | 20 | 200 | 4,000 |
| 2 | 15 | 200 | 3,000 |
| | 273 | | 146,700[a] |

[a]Total cost added to basic vessel quotation.

The attachment of the pipe to the nozzle through which it passes can vary considerably in complexity depending upon the size of the pipe and extent of relative movement that must be allowed. The simplest connection, affording no relative movement, is made by welding the penetrating pipe directly to the nozzle. In this case, the cost for a 12-in. standard-wall carbon-steel pipe connection to the nozzle would be approximately $200. Where relative movement of the pipe with respect to the vessel must be permitted, a bellows expansion joint and reducer are required.

A typical expansion joint penetration is shown in Chapter 9. Table 11.7 presents installed costs for expansion joint penetrations of various diameters and with different capabilities of movement.

Table 11.7. Costs of Expansion Joint Piping Penetrations

| Nozzle Diameter (in.) | Cost ($) Relative to Capability of Movement[a] | |
|---|---|---|
| | 1 in. Axially 1 in. Offset | 2 in. Axially 0 in. Offset |
| 30 | | 1800 |
| 24 | | 1350 |
| 20 | 1300 | |
| 18 | 1080 | 1020 |
| 16 | 850 | 780 |
| 14 | 680 | 660 |
| 12 | 580 | 550 |
| 10 | | 460 |

[a]Costs are contractor's total costs, excluding the nozzle purchased with the containment vessel.

Unit costs of electrical penetrations vary widely because of great differences in penetration design, in the number of penetrations required in a single containment vessel, and in the labor required to install the penetration and make up the wires. The number of individual wires penetrating the containment shell may range from 1000 to 5000 or more. Total installed costs of electrical penetrations range from about $15 per wire for the simple types up to more than $100 per wire for some coaxial cable penetrations. The average total installed unit cost for all electrical penetrations in a typical containment vessel is approximately $30 per wire.

### 11.3.11 Isolation Valves

Isolation valves are required in a containment system to enable the pipes penetrating the containment wall to be closed off rapidly to isolate the containment system in the event of an accident. The isolation valves are of two general types: gate type valves, used primarily in steam and water systems, and butterfly valves, commonly used in containment ventilation systems.

The cost of gate valves varies with pressure rating, material, nondestructive testing required, method of operation, and operating speed. The costs of two types of gate valves are plotted as a function of valve diameter in Figs. 11.8 and 11.9. Costs are shown for both the standard manually operated valve and the quick-closing (10-sec closure) motor-operated isolation valve. The motor-operated isolation valve represented by Fig. 11.8 might be used on main steam lines and reactor feedwater lines. The motor-operated valve on Fig. 11.9 is typical of those used in steam generator feedwater lines. The cost of motor-operated valves increases rapidly as the required closure time is decreased.

Butterfly-type isolation valves are generally used in systems where the pressure is below 150 psig. They have the same cost variables as do gate valves but are usually used where pressure and leakage requirements are less severe and thus they are usually less expensive. The types of butterfly valves and the cost ranges indicated in Table 11.8 are typical of the isolation valves used in containment ventilation systems.

Table 11.8. Cost Ranges of Typical Butterfly Valves

| Type of Valve | Closure Time (sec) | System Pressure (psig) | Diameter Range (in.) | Unit Cost Range ($) |
|---|---|---|---|---|
| Pneumatically operated butterfly valve | 10 | 60 | 4–12 | 500–1800 |
| Motor-operated butterfly valve | 10 | 150 | 12–20 | 1500–2200 |

### 11.3.12 Protective Coatings

Numerous types of coating and paint are used throughout a containment structure. Coatings may be used for corrosion protection, for waterproofing, to permit greater ease in decontamination, or for improved appearance. Approximate installed costs for several representative types of coating used for steel containment structures are presented in Table 11.9.

## 11.4 COST EXPERIENCE ON CONSTRUCTED FACILITIES

The cost experience on constructed plants is often looked to as a reliable indication of the cost that might be incurred in the construction of other plants. Available cost information on containment systems for several nuclear power plants is summarized in Table 11.10. These data were furnished by the Atomic Energy Commission.[19-21]

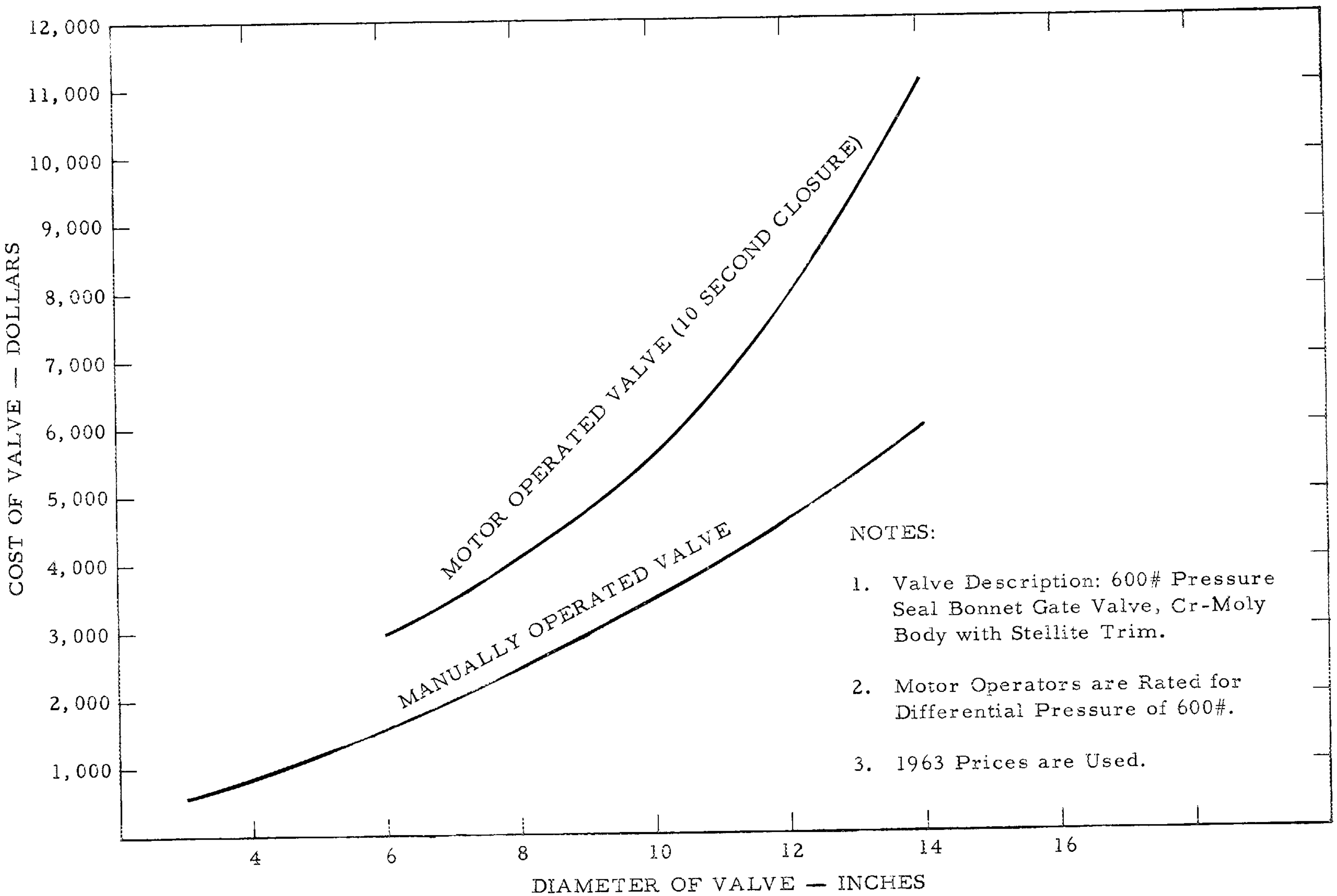


Fig. 11.8. Cost of Gate-Type Isolation Valves as a Function of Valve Diameter.

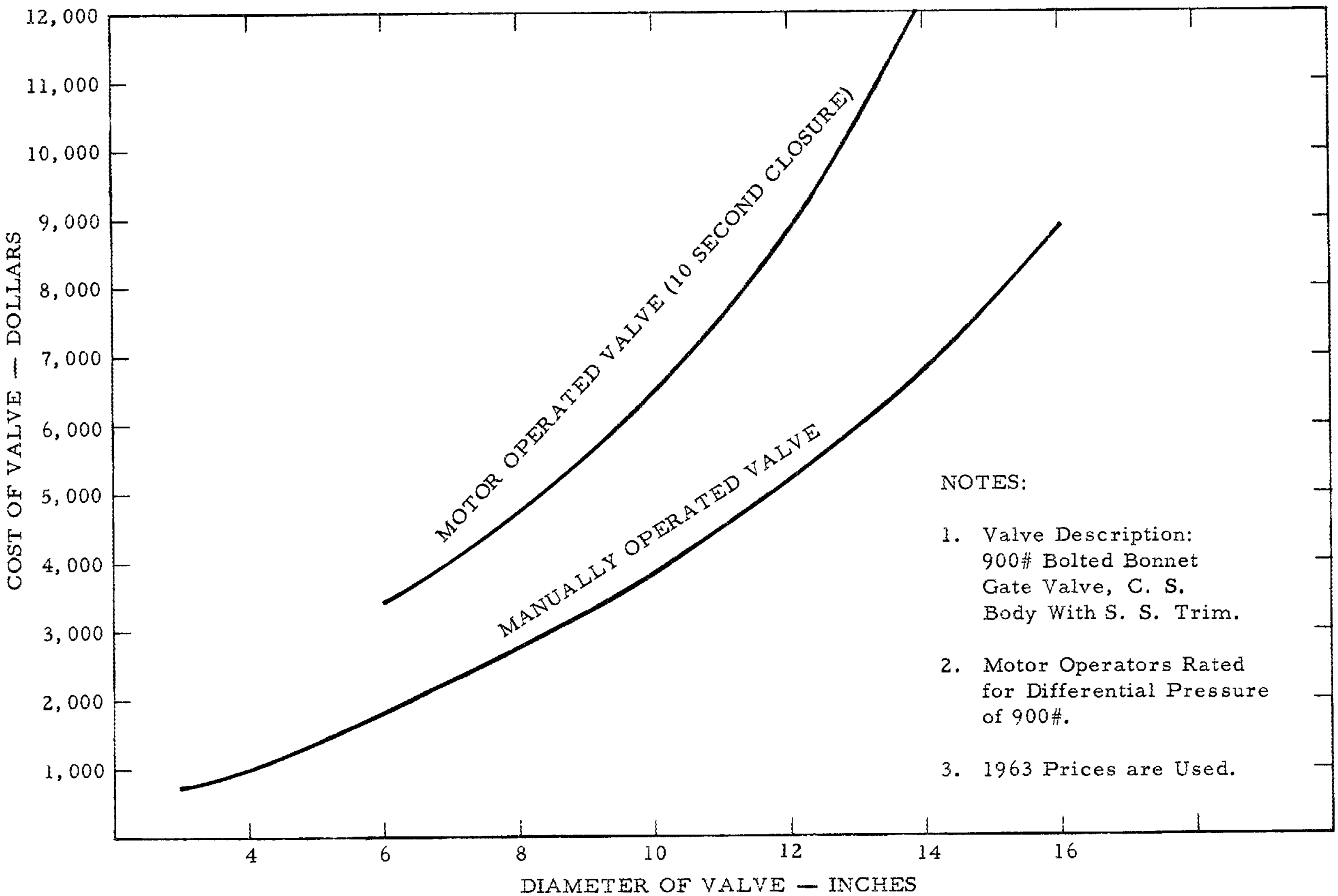


Fig. 11.9. Cost of Gate-Type Isolation Valve as a Function of Valve Diameter.

Table 11.9. Costs of Protective Coatings

| Type of Coating | Application | Unit Cost (including surface preparation) |
|---|---|---|
| Inorganic zinc primer | Applied to the interior exposed surfaces and to the exterior below grade | $0.35/ft$^2$ (cleaning and priming of weld seams, $0.40 to $0.50 per lineal foot) |
| Plastic material | Intermediate and finish coats for interior | $0.10/ft$^2$ for each coat |
| Coal tar epoxy | Applied to exterior over the inorganic zinc primer | $0.12-$0.15/ft$^2$, depending upon amount of cleaning required |
| Zinc chromate primer | Applied in the shop to exterior surface above grade | $0.15/ft$^2$ (cleaning and priming of welds, $0.20 per lineal foot) |
| Alkyd enamel (2 mils thick) | Applied over zinc chromate primer | $0.06-$0.12/ft$^2$, depending on whether one or two coats are needed to build up to required 2-mil thickness |
| Acrylic spray enamel (two coats) | Applied over cork mastic insulation | $0.20-$0.30/ft$^2$ |
| Hypalon (two coats) | Applied over Urethane insulation | ~$0.30/ft$^2$ |

The data of Table 11.10 illustrate the difficulties of attempting to make a meaningful comparison of presumably comparable information or to apply this information to other conditions. Perhaps the most serious inconsistencies in the data arise from the manner in which costs were assigned to each account number. In few, if any, of the cases were costs actually estimated or maintained by using the TID-7025 accounts, and it was necessary to assign costs somewhat arbitrarily to each account after completion of the project. Since this assignment of costs was largely a matter of individual judgment and in most cases there was not sufficient breakdown of costs available to permit assignment to all accounts, it is not surprising that there are wide variations in accounts where greater agreement might be expected. For example, concrete costs as normally tabulated may not distinguish between foundations, shieldings, internal concrete, and external concrete. Thus, shielding may be included in the costs in Table 11.10 in some cases and not in others. Similar qualifications could be made for most of the other accounts.

Table 11.10. Containment Costs for Several Nuclear Reactor Plants[a]

| | Account | EBR-II | Elk River | Hallam | Pathfinder | Piqua | HWCTR | Yankee | EGCR | BONUS |
|---|---|---|---|---|---|---|---|---|---|---|
| 219 | Reactor containment structure | $1,395,000 | | | | | | | | |
| | .1 Excavation and backfill | | $ 7,869 | | $ 102,000 | $ 96,300 | $ 35,000 | $ 11,142 | $ 77,000 | $ 72,462 |
| | .2 Bearing piles and caissons | | | | | 1,200 | | | 133,800 | |
| | .3 Substructure concrete | | 351,885 | | 360,000 | 204,500 | 250,900 | 81,130 | | 556,618 |
| | .4 Superstructure | | | | 625,000 | | | | | |
| | .41 Structural steel | | 8,435 | | | | 79,450 | | | |
| | .42 Containment shell | | 528,917 | | | 320,500 | 322,050 | 1,910,464 | 1,291,000 | 549,697 |
| | .43 Exterior insulation | | 98,139 | | | 58,300 | | | | |
| | .44 Floors, barriers, internal concrete | | 8,690 | | | 82,300 | 358,740 | 997,018 | | |
| | .45 Interior finish | | 109,424 | | | 5,900 | | | | |
| | .6 Building services | | 264,085 | | 62,000 | 122,800 | | 266,515 | 420,000 | 48,131 |
| 212G | Reactor confinement structure | | | | | | | | | |
| | .1 Excavation and backfill | | | $ 128,900 | | | | | | |
| | .2 Bearing piles and caissons | | | | | | | | | |
| | .3 Substructure concrete | | | 2,048,300 | | | | | | |
| | .4 Superstructure | | | 1,206,500 | | | | | | |
| | .5 Stacks (building supported) | | | 37,200 | | | | | | |
| | .6 Building services | | | 939,000 | | | | | | |
| 221.46 | Post incident cooling system | | 44,000 | | | | | | 100,000 | 15,133 |
| 221.5 | Reactor plant containers in the form of tanks installed within a building | 1,387,734 | 1,421,444 | 883,500 | | 53,900 | | | | |
| Total costs chargeable to containment | | $2,782,734 | | $5,243,400 | $1,149,000 | $945,700 | $1,182,714[b] | $3,266,270 | $2,021,800 | $1,242,041 |

[a] This tabulation corresponds with the accounts of the AEC "Guide to Nuclear Power Cost Evaluation," USAEC Report TID-7025 (see ref. 1). The figures presented are the best estimates available from the USAEC. However, because final costs are not yet available in many cases and because the TID-7025 accounts have not been used consistently, these figures should be considered preliminary and not necessarily comparable. Except where noted, costs are only direct costs.

[b] Includes indirect costs.

The wide differences in containment design represented by the plants listed in Table 11.10 are another difficulty. Many of these plants are experimental or developmental and the containment designs are not necessarily "typical" or representative of those that would be built for subsequent plants. Differences in design because of local conditions could also distort the costs and make comparisons of even similar designs difficult. All the containment systems listed in Table 11.10 are described in Chapter 7.

The differences in market conditions and labor rates reflected in the costs but not normally identifiable are a third major difficulty with attempting to compare reported costs. In general, there has been a gradual steady rise in costs (escalation) for the last several years so that, other things being equal, future plants might be expected to cost more than previously constructed plants. However, experience and improved design may tend to offset the general trend for some period of time. Labor rates vary markedly from one location to another. They are generally lowest in the southern part of the U.S. and highest in the Northeast and on the West Coast. Costs of some materials, such as cement and aggregate, vary considerably with location, as well as with general market conditions.

The extent to which these factors enter into the cost presented in Table 11.10 cannot be estimated. It is also difficult to compare these costs with the unit costs presented in Section 11.3, since quantities of materials are not available. Therefore, the information in Table 11.10 may have little quantitative value but may be of interest to indicate magnitudes of costs and general differences in cost distribution among the various designs.

## 11.5 TYPICAL COST EVALUATIONS

### 11.5.1 General

In addition to the fairly specific, but not always consistent, actual cost data presented in Section 11.4, a considerable amount of cost information has been developed in several studies in which the costs of alternate containment systems and containment system components have been estimated and evaluated. Although these studies may present less realistic costs than the actual costs shown in Section 11.4, they are internally consistent and hence more useful for the economic comparison of alternate containment systems. Unfortunately, only a limited number of these examples are available and each of them applies to a rather narrow range of reactor types and sizes. Futhermore, the conclusions of these studies, as with any economic evaluation, are strictly applicable only to the conditions assumed in the ground rules; the results should not be generalized. However, the considerations involved in a thorough economic comparison, if not the actual cost figures, are useful in illustrating the requirements for an evaluation based on other conditions.

## 11.5.2 Reactor Containment Design Study

Probably the most complete economic study of containment structures is that made by Johnson and Nelson.[4,22] In this study, done in two parts, the complete costs of five different containment systems were estimated for each of four different boiling-water reactor plant designs representing three different power ratings and both direct and dual cycles. The containment systems defined for the purpose of the study were as follows:

Scheme I — Standard Containment. This scheme consists of a vapor-tight steel shell designed for internal pressures of 15 to 60 psig. The turbine-generator is not contained within the steel shell.

Scheme II — Pressure-Relief Containment. This scheme is similar to the type employed on the Canadian NPD in which the gas surge from a maximum accident is vented from a low-leakage, low-pressure building by means of a rupture disk. After venting the initial gas release, the vent duct is closed to prevent release of fission products.

Scheme III — Pressure-Suppression Containment. In this scheme a high-pressure primary container (drywell) encloses the reactor and primary system. The steam and water released from the primary system to this container are vented through large pipes into a pool of water that is enclosed in a steel-lined low-pressure concrete chamber (suppression chamber). The water cools the gases and condenses water vapor, resulting in complete containment in a relatively small volume at a low pressure in the suppression chamber.

Scheme IV — Low-Pressure Containment. This scheme consists of a large welded steel hemisphere mounted on a concrete base. The hemisphere contains essentially the entire plant, including the turbine-generator, control room, offices, etc. The structure has sufficient volume to limit the containment pressure to less than 5 psig.

Scheme V — Total Containment. This scheme is similar to low-pressure containment in that the entire plant is contained, but it is not limited to a design pressure of 5 psig, so a more compact design is possible. This concept was added in the supplemental report, ref. 4.

The four reactor plants considered were the following:

1. 44-Mw(e), direct-cycle, natural-circulation, boiling-water reactor plant,
2. 180-Mw(e), dual-cycle, forced-circulation, boiling-water reactor plant,
3. 300-Mw(e), dual-cycle, forced-circulation, boiling-water reactor plant,
4. 300-Mw(e), direct-cycle, forced-circulation, boiling-water reactor plant with internal steam separation.

Preliminary design data were established for each containment scheme and each plant type, as listed in Tables 11.11 and 11.12. The items that changed from case to case were identified and cost differences for these components were determined. Hence, this study was an attempt to establish the total change in plant cost corresponding to a change in containment concept. The results of this study are summarized in Table 11.13, which gives containment costs both in \$/kw(e) and in mills/kwhr for each

Table 11.11. Containment Study Design Data[a]

| | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Electrical rating, Mw | 44 | | | | 180 | | | | 300 | | | |
| Thermal rating, Mw | 133 | | | | 630 | | | | 980 | | | |
| Reactor system pressure, psia | 965 | | | | 1,015 | | | | 990 | | | |
| Reactor system temperature, °F | 540 | | | | 546 | | | | 543 | | | |
| Steam volume, $ft^3$ | 1,217 | | | | 4,720 | | | | 5,810 | | | |
| Water volume, $ft^3$ | 1,002 | | | | 8,082 | | | | 9,054 | | | |
| Total weight of steam and water, lb | 57,600 | | | | 389,292 | | | | 443,000 | | | |
| Containment scheme[b] | I | II | III | IV | I | II | III | IV | I | II | III | IV |
| Containment configuration | Cylinder | Rectangular | Cylinder | Hemisphere surmounting cylinder | Sphere | Rectangular | Sphere | Hemisphere surmounting cylinder | Sphere | Rectangular | Sphere | Hemisphere surmounting cylinder |
| Containment building diameter, ft | 64 | | 17 | 204 | 190 | | ~190 | 400 | 190 | | ~190 | 412 |
| Containment volume, gross, $ft^3$ | $386 \times 10^3$ | $275 \times 10^3$ | $12.7 \times 10^3$ | $3,100 \times 10^3$ | $3,591 \times 10^3$ | $1,200 \times 10^3$ | $\sim 3,600 \times 10^3$ | $20,900 \times 10^3$ | $3,591 \times 10^3$ | $1,600 \times 10^3$ | $\sim 3,600 \times 10^3$ | $23,800 \times 10^3$ |
| Containment volume, net, $ft^3$ | $243 \times 10^3$ | $226 \times 10^3$ | $9.62 \times 10^3$ | $2,480 \times 10^3$ | $2,860 \times 10^3$ | $960 \times 10^3$ | $\sim 2,900 \times 10^3$ | $16,700 \times 10^3$ | $2,792 \times 10^3$ | $1,280 \times 10^3$ | $\sim 2,800 \times 10^3$ | $19,000 \times 10^3$ |
| Containment design pressure, psig | 40 | 5 | Cylinder, 55 end domes, 150 | 5 | 23 | 5 | 15 | 5 | 28 | 5 | 15 | 5 |
| Containment wall thickness, in. | Top, 047 Cylinder and bottom, 0.94 | 16 | Cylinder, 0.35 Top, 1.35 Bottom, 2.00 | 5/16 | 0.87–1.23 | 24 | 0.57–1.15 | 5/16 | 0.94–1.25 | 24 | 0.57–1.15 | 5/16 |
| Containment wall material | Steel | Concrete | Steel | Steel and concrete | Steel | Concrete | Steel | Steel and concrete | Steel | Concrete | Steel | Steel and concrete |

[a]From ref. 22, p. 18.

[b]Scheme I – Standard Containment.
Scheme II – Pressure-Relief Containment.
Scheme III – Pressure-Suppression Containment.
Scheme IV – Low-Pressure Containment.

Table 11.12. Containment Study Design Data[a]

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| Cycle | Direct | Dual | Dual | Direct | | | | |
| Electrical rating, Mw | 44 | 180 | 300 | 300 | | | | |
| Thermal rating, Mw | 133 | 630 | 980 | 871 | | | | |
| Reactor system pressure, psia | 965 | 1,015 | 990 | 1,250 | | | | |
| Reactor system temperature, °F | 540 | 546 | 543 | 572 | | | | |
| Steam volume, $ft^3$ | 1,217 | 4,720 | 5,810 | 1,935 | | | | |
| Water volume, $ft^3$ | 1,002 | 8,082 | 9,054 | 1,532 | | | | |
| Total weight of steam and water, lb | 57,600 | 389,292 | 443,000 | 199,850 | | | | |
| Containment scheme[b] | V | V | V | I | II | III | IV | V |
| Containment configuration | Hemisphere surmounting cylinder | Hemisphere surmounting cylinder | Hemisphere surmounting cylinder | Cylinder | Rectangular | Cylinder | Hemisphere surmounting cylinder | Hemisphere surmounting cylinder |
| Containment building diameter, ft | 130 | 230 | 240 | 84 | | Top, 21<br>Bottom, 29 | 286 | 200 |
| Containment volume, gross, $ft^3$ | $1,280 \times 10^3$ | $5,420 \times 10^3$ | $6,510 \times 10^3$ | $908.4 \times 10^3$ | $525 \times 10^3$ | $34.1 \times 10^3$ | $9,020 \times 10^3$ | $3,670 \times 10^3$ |
| Containment volume, net, $ft^3$ | $1,030 \times 10^3$ | $4,340 \times 10^3$ | $5,210 \times 10^3$ | $750 \times 10^3$ | $420 \times 10^3$ | $25.3 \times 10^3$ | $8,400 \times 10^3$ | $2,940 \times 10^3$ |
| Containment design pressure, psig | 11 | 17 | 16 | 49 | 5 | 127 | 5 | 14 |
| Containment wall thickness, in. | Hemisphere, 0.26<br>Cylinder, 0.52 | Hemisphere, 0.71<br>Cylinder, 1.42 | Hemisphere, 0.70<br>Cylinder, 1.40 | Top, 0.75<br>Cylinder, 1.50<br>Bottom, 1.50 | Upper wall, 18<br>Lower wall, 24 | Top, 0.75<br>Upper cylinder, 1.0<br>Lower cylinder, 1.35<br>Bottom, 1.375 | Hemisphere, 0.26<br>Cylinder, 0.52 | Hemisphere, 0.59<br>Cylinder, 1.18 |
| Containment materials | Steel and concrete | Steel and concrete | Steel and concrete | Steel | Concrete | Steel | Steel and concrete | Steel and concrete |

[a] From ref. 4, p. 14.

[b] Scheme I — Standard Containment
Scheme II — Pressure-Relief Containment
Scheme III — Pressure-Suppression Containment
Scheme IV — Low-Pressure Containment
Scheme V — Total Containment

Table 11.13. Containment Cost Comparison[a]

| Plant Electrical Rating (Mw) | Plant Cycle | Scheme I — Standard Containment | Scheme II — Pressure Relief[b] | Scheme III — Pressure Suppression[b] | Scheme IV — Low Pressure[b] | Scheme V — Total Containment[b] |
|---|---|---|---|---|---|---|
| | | | Costs in \$/kw(e) | | | |
| 44 | Direct | 31.05 | 12.90 | 17.86 | 71.59[c] | 26.83[c] |
| 180 | Dual | 30.63 | 6.46 | 38.59 | 88.36 | 28.19 |
| 300 | Dual | 21.07 | 4.62 | 25.14 | 59.35 | 20.73 |
| 300 | Direct | 8.91 | 2.57 | 4.80 | 19.10[c] | 10.35[c] |
| | | | Costs in mills/kwhr | | | |
| 44 | Direct | 0.63 | 0.26 | 0.36 | 1.46[c] | 0.55[c] |
| 180 | Dual | 0.63 | 0.13 | 0.79 | 1.80 | 0.58 |
| 300 | Dual | 0.43 | 0.09 | 0.51 | 1.21 | 0.42 |
| 300 | Direct | 0.18 | 0.05 | 0.10 | 0.39[c] | 0.21[c] |

[a]From ref. 4, p. 33.

[b]Costs include adjustments for differences in piping, electrical installation, fuel handling, etc., as well as differences in the containment structure itself.

[c]Cost of moat not included.

scheme. It should be pointed out that the Scheme V costs include costs of housing the turbine, generator, condenser, etc., whereas the costs for the other four schemes do not.

The containment designs employed in this study are, in general, quite representative of their type and, perhaps with the exception of pressure relief, equivalent in their ability to contain radioactivity in the event of the maximum accident, which is taken as a complete break in the largest primary system pipe, with subsequent release of fission products from the core. There are, however, substantial variations in the direct radiation dose at the site boundaries and the extent to which the design is susceptible to missile damage. Chapters 1 and 7 discuss general and specific applications of these various designs and their relative technical merits. Also refs. 22 and 4 contain fairly detailed design descriptions and cost breakdowns for each of the cases considered. These should be consulted to determine the applicability of the cost comparisons to a particular situation. Because of the detailed information presented, it should be possible to adjust the costs to apply to somewhat different conditions if necessary.

### 11.5.3 Selection of the Containment Structure for the Dresden Nuclear Power Station

Smith and Randolph[23] have discussed some of the studies performed in connection with the selection of a containment structure for the Dresden Nuclear Power Station. Eight containment schemes were considered. Four were "total" plant containment arrangements, in which essentially the entire plant, including the turbine-generator, was enclosed within the containment structure, and four were so-called "partial" containment arrangements, in which the turbine-generator, condenser, etc., were outside the containment structure.

The results of the economic evaluations made in this study are compared on a relative cost basis in Table 11.14. The costs are given in percentage of the cost of the 200-ft sphere for total containment. The costs for partial containment include the cost for a separate turbine

Table 11.14. Summary of Dresden Containment Structure Study[a]

| Containment Design | Relative Cost (%) (200-ft sphere = 100%) |
|---|---|
| "Total" containment | |
| 200-ft-diam steel sphere, 75% above ground; design pressure, 22.4 psig; wall thickness, 1.0 to 1.25 in. (reference case) | 100 |
| 220-ft-diam buried cylindrical vault; 110 ft high; rock cavity lined with concrete and faced with a metallic membrane; design pressure, 22.0 psig | 137 |
| 220-ft-diam partially buried cylinder similar to above but only one-third below ground | 115 |
| 200-ft diam steel-lined concrete cylinder with steel hemispherical dome top; cylindrical portion largely below ground; design pressure, 22.0 psig | 98 |
| "Partial" containment | |
| 170-ft-diam steel sphere similar to 200-ft-diam reference design; design pressure, 36.3 psig; wall thickness, 1.37 in. | 97 |
| 110-ft-diam capsule vault; vertical steel cylinder with hemispherical ends buried in rock and encased in concrete; design pressure, 76.3 psig; wall thickness, 0.5 in. | 120 |
| 110-ft-diam partially buried capsule similar to above but only 90 ft is below grade; design pressure, 61.7 psig; wall thickness, 1.5 in. | 101 |
| 180-ft-diam cylinder with hemispherical dome top; similar to 200-ft dome-top design | 103 |

[a]From Smith and Randolph, ref. 23.

building. The effect of congestion on construction within the containment structures was included as a penalty. From this evaluation, it was possible to rule out the buried cylindrical vault, the capsule vault, and the partially buried cylinder. The other five configurations were essentially equal within the limits of accuracy of the estimates. The concept represented by the 170-ft-diam sphere employing partial containment was selected because it had a greater available margin for accommodating future design changes and because it was capable of being designed, constructed, and tested as a Code vessel. Later design changes in the plant led to an increase in the size of the containment vessel to a 190-ft-diam sphere.

### 11.5.4 Cost Normalization Study

In 1959, the Atomic Energy Commission sponsored a study for normalizing the costs of various nuclear power plant designs that had been prepared for the Commission by various contractors. The results of this normalization study have been reported.[2] Eight types of reactor were considered at each of three electrical power ratings: 75, 200, and 300 Mw. In most cases, detailed analyses were made for the 200-Mw plants and the costs were extrapolated to the other sizes. However, for the organic-cooled and the sodium-graphite plants the base case was 75 Mw and for the fast breeder reactor a base size of 150 Mw was used.

The following types of plant were included in the study:

1. Pressurized-water reactor,
2. Boiling-water reactor,
3. Organic-cooled reactor,
4. Sodium-cooled graphite-moderated reactor,
5. Sodium-cooled fast reactor,
6. Aqueous homogeneous reactor,
7. Heavy-water-moderated reactor,
8. Gas-cooled reactor.

The plant designs considered were representative of the state-of-the-art for that period and thus contained designs for some plants that were proven and conservative and others that were quite speculative. The containment system for each case was that which had been adopted by the principal proponent of the reactor type. The study presents a brief capital cost breakdown for each case, including a cost for the reactor containment structures.

Table 11.15 gives containment structure and total plant cost data extracted from this reference. Since this study was concerned with the total plant cost rather than with containment cost, the containment cost was not broken down in further detail. Engineering and other indirect costs were determined as a percentage of direct capital cost.

The containment system employed for each plant was as follows:

1. The reference pressurized-water reactor system was a 200-Mw(e) plant employing a 135-ft-diam steel sphere containing the reactor, reactor auxiliary equipment, and steam-generating facilities.

2. The reference boiling-water reactor system was a 200-Mw(e) dual-cycle plant. The reactor, reactor auxiliary equipment, and secondary

Table 11.15. Capital Cost Breakdown for Various Reactor Plant Types[a]

| | Cost Based on Electrical Output | | |
|---|---|---|---|
| | 75 Mw(e) | 200 Mw(e) | 300 Mw(e) |
| Pressurized-water reactor plants | | | |
| Reactor container structures | \$ 2,480,000 | \$ 2,540,825 | \$ 3,100,000 |
| Total direct construction costs[b] | 23,724,045 | 41,037,940 | 53,479,575 |
| Total capital cost | \$32,644,445 | \$56,412,734 | \$ 73,419,035 |
| Total capital cost, \$/kw(e) | 435 | 282 | 242 |
| Boiling-water reactor plants | | | |
| Reactor container structures | \$ 3,000,000 | \$ 5,865,100 | \$ 7,050,000 |
| Total direct construction costs[b] | 25,622,364 | 45,275,480 | 57,503,760 |
| Total capital cost | \$35,249,404 | \$62,202,728 | \$ 78,927,740 |
| Total capital cost, \$/kw(e) | 470 | 311 | 263 |
| Organic-cooled reactor plants | | | |
| Reactor container structures | \$ 2,302,700 | \$ 3,961,450 | \$ 4,104,450 |
| Total direct construction costs[b] | 18,917,785 | 34,754,610 | 47,595,510 |
| Total capital cost | \$26,237,563 | \$48,218,071 | \$ 65,978,161 |
| Total capital cost, \$/kw(e) | 350 | 241 | 220 |
| Sodium-cooled graphite-moderated reactor plants | | | |
| Reactor container structures | \$ 4,568,200 | \$ 7,350,000 | \$ 9,300,000 |
| Total direct construction costs[b] | 30,725,980 | 52,088,636 | 65,838,420 |
| Total capital cost | \$42,466,178 | \$71,994,496 | \$ 90,910,462 |
| Total capital cost, \$/kw(e) | 565 | 360 | 303 |
| Fast sodium-cooled reactor plants | | | |
| Reactor container structure | \$ 2,138,700 | \$ 4,138,700[c] | \$ 4,658,700 |
| Total direct construction costs[b] | 24,694,930 | 36,902,505[c] | 55,377,464 |
| Total capital cost | \$34,102,220 | \$50,998,356[c] | \$ 76,485,014 |
| Total capital cost, \$/kw(e) | 455 | 340[c] | 255 |
| Aqueous homogeneous reactor plants | | | |
| Reactor container structures | \$ 3,700,000 | \$ 8,349,200 | \$ 10,845,000 |
| Total direct construction costs[b] | 21,973,100 | 46,783,040 | 62,111,300 |
| Total capital cost | \$33,775,410 | \$72,298,044 | \$ 96,862,530 |
| Total capital cost, \$/kw(e) | 450 | 362 | 323 |
| Heavy-water-moderated reactor plants | | | |
| Reactor container structures | \$ 4,250,200 | \$ 4,878,600 | \$ 5,134,000 |
| Total direct construction cost[b] | 29,950,130 | 49,775,370 | 62,320,120 |
| Total capital cost | \$47,995,740 | \$84,997,407 | \$107,999,930 |
| Total capital cost, \$/kw(e) | 640 | 425 | 360 |
| Gas-cooled reactor plants | | | |
| Reactor container structures | \$ 4,930,000 | \$ 5,804,800 | \$ 6,680,000 |
| Total direct construction costs[b] | 36,878,455 | 66,038,400 | 83,333,091 |
| Total capital cost | \$50,585,605 | \$90,500,440 | \$114,127,001 |
| Total capital cost, \$/kw(e) | 675 | 452 | 380 |

[a] From ref. 2.

[b] Direct construction cost estimates include construction, contractors' field office expense, tools, construction equipment, overhead, and profit.

[c] Calculations for a 150-Mw(e) plant.

steam generating facilities were contained within a 190-ft-diam steel sphere.

3. The reference organic-cooled reactor system was a 75-Mw(e) indirect-cycle plant. The reactor, reactor auxiliary equipment, and steam-generating facilities were located in a 140-ft-diam cylindrical concrete structure 60 ft high. This cylinder was topped by a 136-ft-diam hemispherical steel dome.

4. The reference sodium-cooled graphite-moderated reactor system was a 75-Mw(e) plant. The reactor, reactor auxiliary equipment, and secondary steam-generating facilities were located in an insulated, rectangular, metal-panel building approximately 275 ft long, 152 ft wide, and 75 ft high. All the primary equipment was located below grade in thick-walled concrete vaults.

5. The reference sodium-cooled fast reactor plant had a net electrical output of 150 Mw(e). The reactor, reactor auxiliary equipment, and intermediate heat exchangers were located within a cylindrical steel container 72 ft in diameter and 120 ft high. The container had a hemispherical top head and a dished lower head.

6. The reference aqueous homogeneous reactor plant was a 200-Mw(e) system having two reactors. Each reactor and primary system was located within a cylindrical steel building 66 ft in diameter and 116 ft high; the cylinders had dished heads. The buildings were approximately one-half below grade and designed so that they could be completely filled with water to permit maintenance on the systems.

7. The reference heavy-water-moderated reactor system was a 200-Mw(e) plant with the reactor, reactor auxiliary equipment, and steam-generating equipment all located within a 164-ft-diam spherical steel container.

8. The reference gas-cooled reactor system was a 200-Mw(e) plant. The reactor was housed within a 12-ft-thick ordinary concrete enclosure that comprised the reactor shield and building. The steam generators were outdoors and the primary system blower equipment was located in enclosures attached to the main building. The reactor pressure vessel was 70 ft in diameter with 3-in.-thick carbon steel walls. The building was more or less octagonal in plan and closely conformed to the reactor vessel.

The rapid progress in power reactor technology and economics since 1959, when this study was made, has resulted in a number of recent plant price quotations that are substantially lower than the costs developed in this study. This is to be expected as more experience is gained and as more plants are built for commercial power production.

### 11.5.5 Economic Aspects of Reactor Safety

Kallman and Hanson[16] have discussed some economic aspects of containment specifically related to a 500-Mw(e) pressurized-water reactor plant. In their study they evaluated some of the costs associated with containment requirements for the plant in a semipopulous area, including extra costs for auxiliary systems and for power transmission lines required because of nuclear hazards considerations.

The containment vessel considered was a 160-ft-diam steel sphere with a 7/8-in. wall thickness. It incorporated a concrete shield or "igloo" completely surrounding the aboveground portion of the container to permit operation at adjacent facilities in the event of an accident that might fill the sphere with radioactive gases. They estimated the costs listed in Table 11.16.

Table 11.16. Costs of Special Nuclear Safety Equipment[a]

| Equipment | Cost |
| --- | --- |
| Reactor container, including excavation, foundation, ventilation and spray systems, air locks, and penetrations | \$3,800,000 |
| Concrete "Igloo" | 1,200,000 |
| Extra transmission line, estimated equivalent to 30 miles at \$67,000 per mile | 2,000,000 |
| Extra auxiliary equipment | 1,400,000 |
| | \$8,400,000 |

[a]From Kallman and Hanson, ref. 16.

They also estimated that a conventional building 200 ft long, 120 ft wide, and 60 ft high would be required to enclose the reactor if there were no containment provisions and that the building would cost \$3,000,000. Thus, the special features, including containment, listed in Table 11.16 for protection against nuclear hazards actually add about \$5,400,000 to the plant cost, which is in the range of \$20 to \$35 per kw(e).

### 11.5.6 Steel Containment Vessel Costs

The Oak Ridge National Laboratory[24] considered the cost of steel containment vessels in a study performed in 1959. In this study the containment vessel installed cost was correlated with vessel wall thickness and diameter for both spherical and cylindrical vessels. Figures 11.10 through 11.17 (taken from ref. 24) show the correlations that were developed. These curves are based on assumed average steel and fabrication costs for bare steel shells. They may be used for making rough first estimates of steel shell costs, including the cost of excavation and backfill and of external foundation concrete. These same curves are reproduced in Volume 2 of the AEC cost evaluation guide.[1]

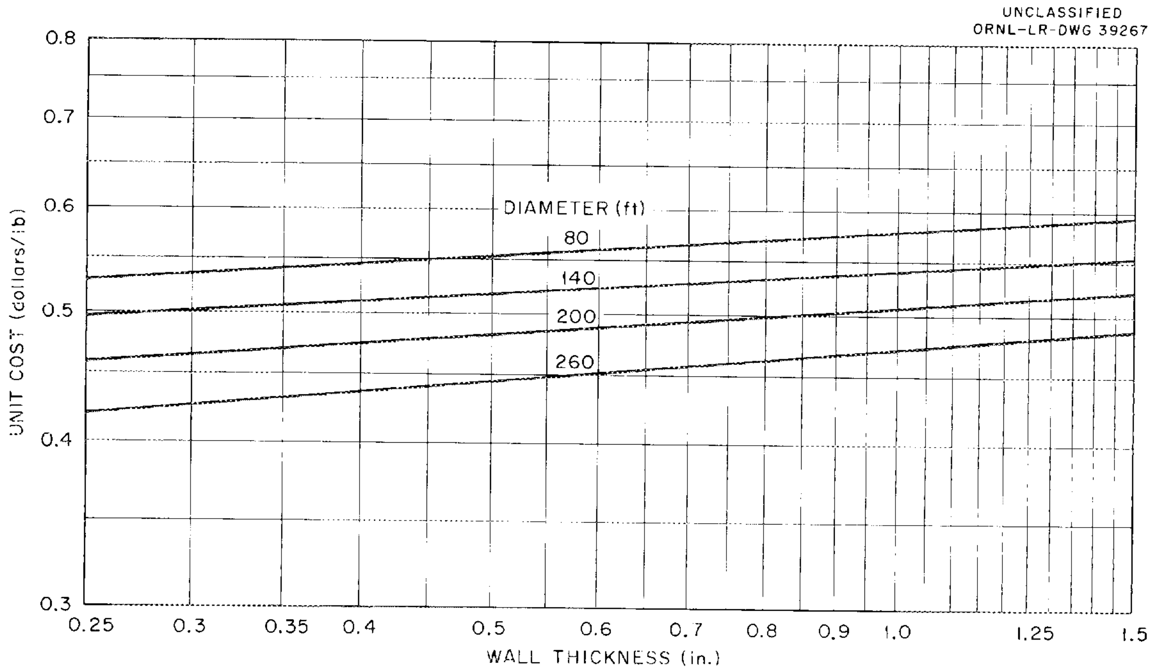


Fig. 11.10. Spherical Containment Vessel Unit Cost as a Function of Diameter and Wall Thickness.

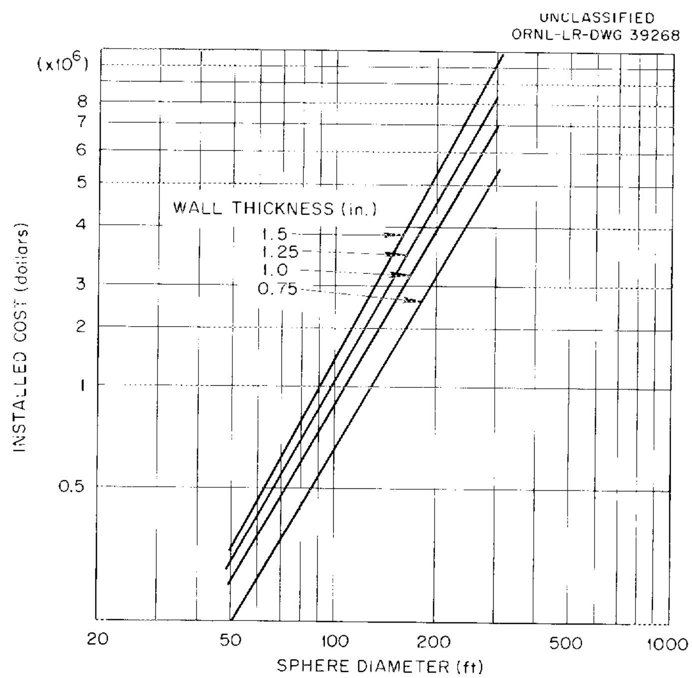


Fig. 11.11. Approximate Installed Cost of a Spherical Containment Vessel as a Function of Diameter and Wall Thickness.

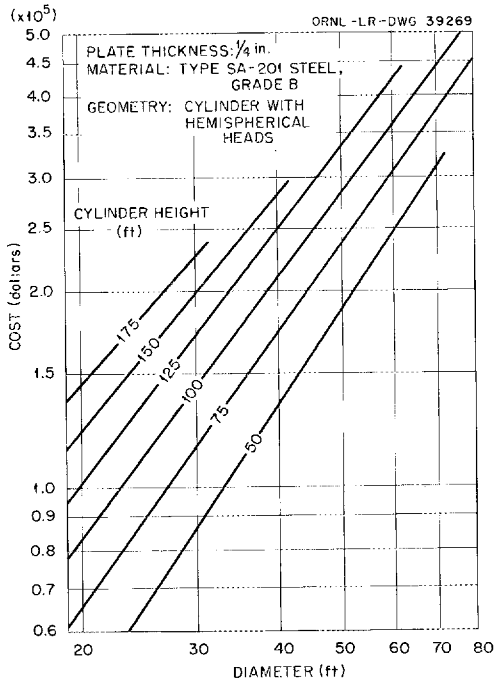


Fig. 11.12. Approximate Cost of Cylindrical Containment Vessels Fabricated from 1/4-in.-Thick Plate.

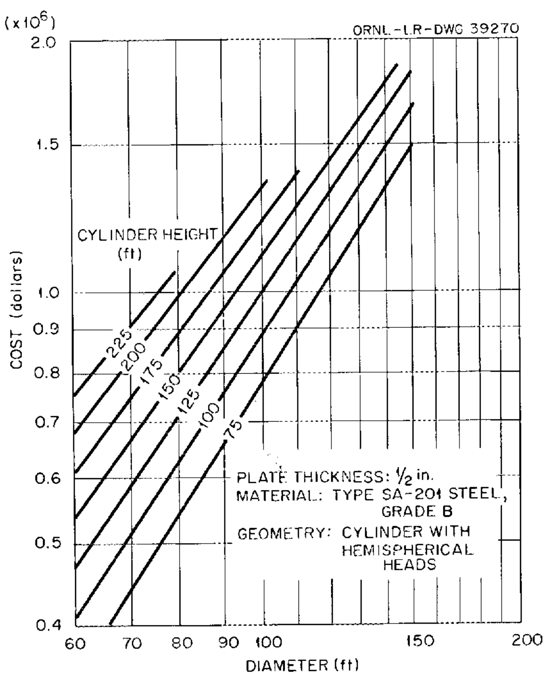


Fig. 11.13. Approximate Cost of Cylindrical Containment Vessels Fabricated from 1/2-in.-Thick Plate.

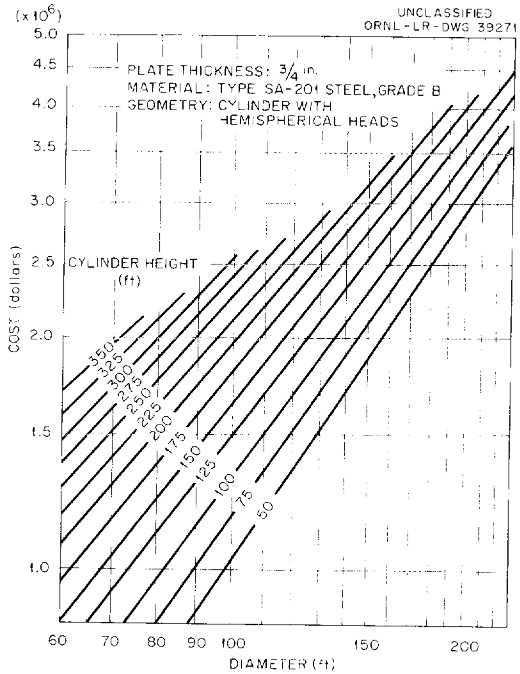


Fig. 11.14. Approximate Cost of Cylindrical Containment Vessels Fabricated from 3/4-in.-Thick Plate.

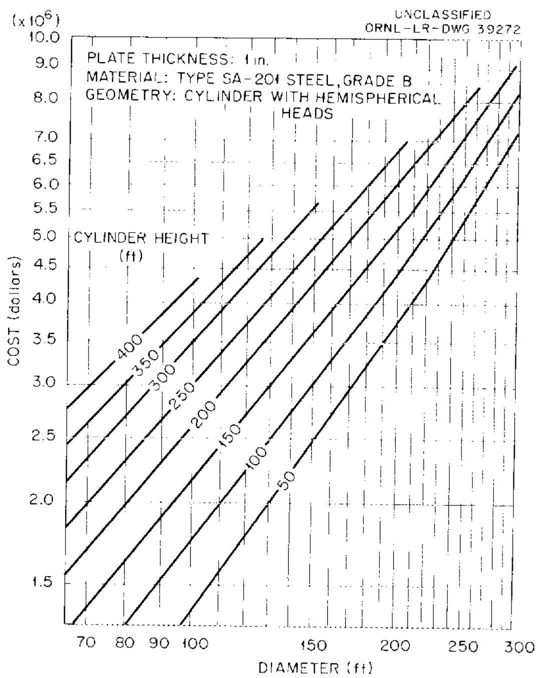


Fig. 11.15. Approximate Cost of Cylindrical Containment Vessels Fabricated from 1-in.-Thick Plate.

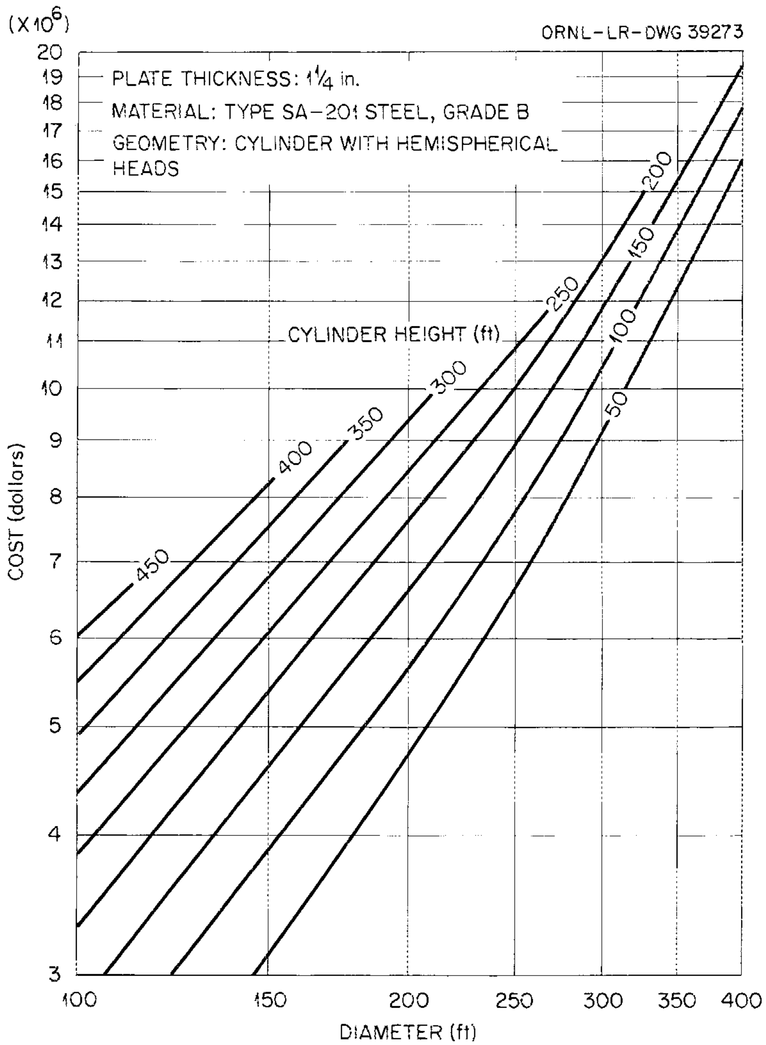


Fig. 11.16. Approximate Cost of Cylindrical Containment Vessels Fabricated from 1 1/4-in.-Thick Plate.

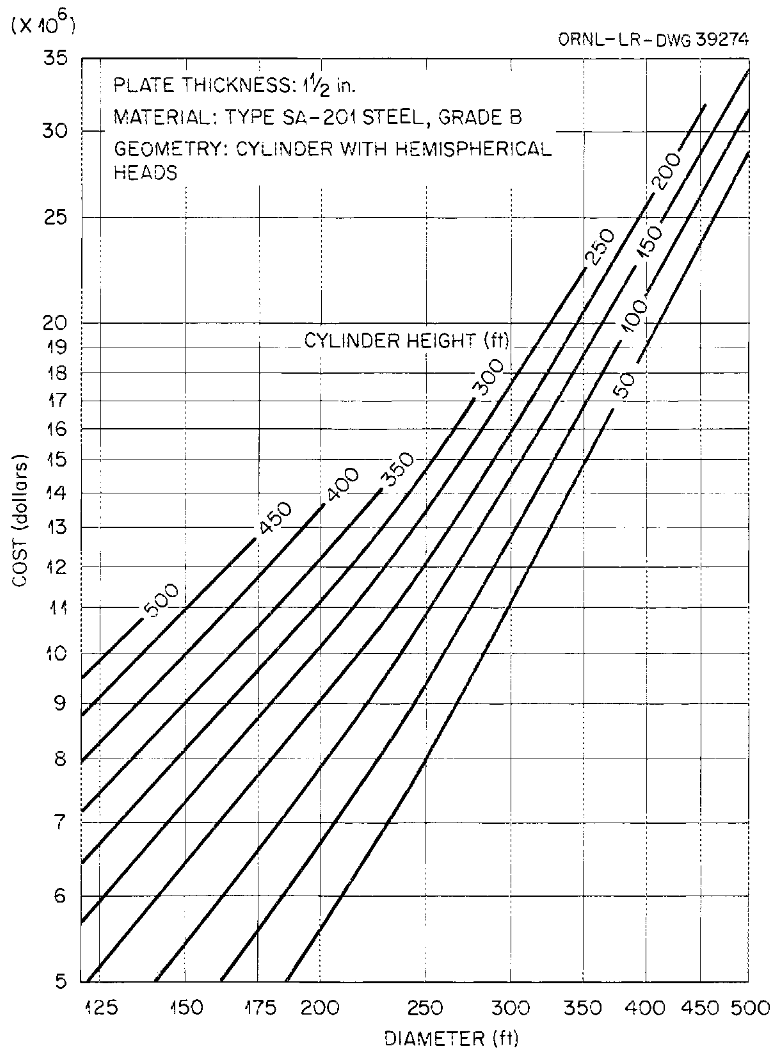


Fig. 11.17. Approximate Cost of Cylindrical Containment Vessels Fabricated from 1 1/2-in.-Thick Plate.

### 11.5.7 Conventional Building Costs

Koontz and his co-workers[25] performed tests on various components of low-pressure, low-leakage buildings similar in construction to conventional industrial-type buildings. These types of buildings have economic advantages over conventional containment structures for those situations where they can be used. Table 11.17, taken from ref. 25, indicates the relative costs of five different types of wall construction evaluated in the study. Except for Type 1 (designed for a pressure of 15 in. $H_2O$), all the walls are designed for a pressure differential of 5 psig. The range of cost variation of the last four structures is only about 20%. Additional studies[26] have been made of the cost of improving conventional buildings to withstand low accident pressures and to achieve low leakages. The costs of reducing leakage were found to be relatively small and, for some accidents, the improved conventional building appeared to have some economic advantage over steel shell containment.

### 11.5.8 Underground Containment Cost Study

Beck[27] has reported on a study of underground containment in which a preliminary cost estimate was developed for an underground version of the 5-Mw(e) Experimental Boiling Water Reactor (EBWR). The reactor and service building (containing the entire plant, exclusive of the cooling tower) were replaced by two adjacent chambers having the exact size and shape as their counterparts in the plant as constructed. The chamber surfaces were covered by 12 to 14 in. of concrete and lined with light-gage metal. It was estimated that this underground design would add about $322,000, or approximately 7%, to the cost of the facility.

Beck also presented some unit costs for rock excavation and associated features. He indicated that the unit costs in the U.S. may vary from $10 to $30 per cubic yard, whereas in Sweden, where labor rates are lower and equipment and techniques are quite advanced, the costs might range from $3.20 to $15 per cubic yard. For the estimate on the underground EBWR, Beck used the unit costs shown in Table 11.18.

### 11.5.9 Filter System Costs

The costs of air and gas cleaning systems as applied in nuclear installations were discussed by First and Silverman in an article in Nuclear Safety.[28] An approximate relationship between installed cost and efficiency of gas cleaning systems that was reproduced in the Nuclear Safety article from a previous paper by Silverman[29] is shown in Fig. 11.18. Obviously, any such relationship cannot take into account all significant variables, such as the cost of in-place testing, capacity of the system, size and type of particles to be removed, concentration of particles in the gas stream, and assignment of costs between the gas cleaning portions and the conventional heating and ventilating or gas handling portions of the system. Furthermore the data shown are only for filtration of dust and do not include the cost of special filters, such as charcoal absorbers, caustic scrubbers, etc. Nevertheless, the number of practical devices that can be used for specific gas cleaning applications is limited, and the correlation presented in Fig. 11.18 considers most of the devices that can be used for gas cleaning of inert dusts.

Table 11.17. Relative Costs of Five Different Types of 40- by 40-ft Walls[a,b]

| | Type 1, Metal Panel, Steel Framing | Type 2, Metal Panel, Reinforced Gunite Concrete with Steel Framing | Type 3, Reinforced Concrete (Tiltup) with Steel Framing | Type 4, Prestressed Concrete with Steel Framing | Type 5, Prestressed Concrete with Steel and Prestressed Concrete Framing |
|---|---|---|---|---|---|
| Wall cost | $2,994 | $ 6,565 | $ 4,480 | $ 5,692 | $ 5,692 |
| Framing cost | 1,079 | 5,716 | 5,716 | 5,716 | 6,464 |
| Foundation cost | | 181 | 222 | 181 | 264 |
| Totals | $4,073 | $12,462 | $10,148 | $11,589 | $12,420 |

[a]Types 2 through 5 designed to withstand a pressure differential of 5 psi; type 1 designed to withstand a pressure differential of 15 in. $H_2O$. Costs should be compared on a relative basis only. Unit costs are based on a complete building construction contract, not just the 40- by 40-ft wall.

[b]From Koontz et al., ref. 25, p. V-6.

Table 11.18. Assumed Unit Costs for Underground Containment Construction[a]

| Item | Unit Cost ($\$/yd^3$) |
|---|---|
| Excavation | |
| Large chambers | 15 |
| Tunnels and small chambers | 25 |
| Concrete lining | |
| 24-in. walls and roof | 52 |
| Floors | 30 |
| 12-in. walls and roof | 61 |

[a]Based upon 14% escalation of Bureau of Mines 1954 costs in "Manual on Design of Underground Installations in Rock," p. 11-32.

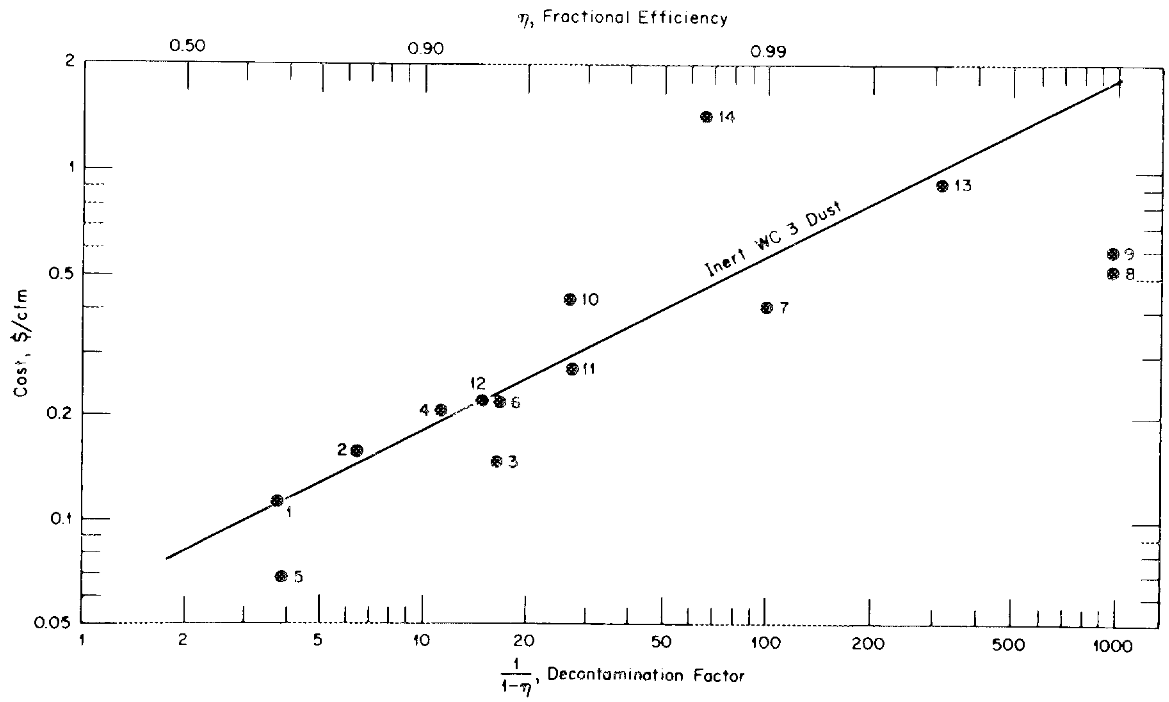


Fig. 11.18. Cost vs Decontamination Factor for Gas Cleaning of Inert Dusts. (From First and Silverman, ref. 28)

This article also considers total annual costs and the breakdown of these costs between capitalization, power, replacement filters, and labor for both ultrahigh-efficiency filters and throwaway prefilters. This information is summarized in Table 11.19. As indicated, ultrahigh-efficiency filters cost seven to eight times as much as the prefilters. For both types, filter replacement and maintenance costs represent 60 to 70% of the total annual costs, with the cost of replacement filters the major cost for the ultrahigh-efficiency filter system and labor costs most significant for the prefilters. This reflects the much greater purchase cost for the ultrahigh-efficiency filters, which is generally more than 25 times the cost of the prefilter units. The abnormally high percentage capitalization cost for the 100,000-cfm ultrahigh-efficiency filters is due to the fact that the systems of this size that were evaluated all operated at well below rated capacity (20 to 33%).

## 11.6 COST ESTIMATING

### 11.6.1 General

Previous sections, in presenting cost information for various types of containment systems, have pointed out some of the necessary considerations in attempting to apply these costs to different situations or even to compare the costs of similar designs under seemingly similar conditions. The conclusions that might be reached in any such economic evaluation could be quite misleading, if not entirely erroneous, if they were not based on a thorough understanding of the particular conditions involved and on the many factors other than the containment design that could have a strong bearing on the economics of the situation. This section describes in general terms the basic steps required in preparing a cost estimate for any facility and attempts thereby to indicate the problems involved in attempting to present containment cost information that could be applied to a particular situation. Also presented in this section, as an illustration of a cost estimate of a particular plant, is the detailed cost estimate of the HWCTR containment system.

### 11.6.2 General Cost Estimating Procedure

The degree of accuracy desired in a cost estimate, and hence the degree of detail and the amount of effort required, depends on the purpose for which the estimate is to be used. Estimates may be prepared for many different purposes, such as to study the economic feasibility of a project, to provide a basis for appropriation or allocation of funds for a proposed project, for budgetary purposes, to provide the basis for an economic comparison of alternate designs, or to provide the basis for a firm price bid. There are differences in approach between estimates prepared for these different purposes, and the nature of the estimate will be determined by the purpose for which it is intended. However, the principal steps followed in preparing any estimate are generally the same, differing

Table 11.19. Total Annual Cost and Cost Breakdown for Filter Systems[a]

| Total Capacity (cfm) | Number of Units in Survey | Total Fixed Plus Operating Cost [$/(1000 cfm) (year)] | | | Percentage of Total Annual Unit Costs | | | |
|---|---|---|---|---|---|---|---|---|
| | | Average | Minimum | Maximum | Capitalization | Power | Replacement Filters | Labor |
| | | | | Ultrahigh-Efficiency Filters | | | | |
| 100 | 4 | 935 | 175 | 2,678 | 17.6 | 8.2 | 64.0 | 10.3 |
| 1,000 | 22 | 359 | 152 | 1,227 | 15.5 | 14.7 | 60.3 | 9.5 |
| 10,000 | 10 | 175 | 152 | 216 | 11.5 | 19.4 | 61.2 | 8.6 |
| 100,000 | 3 | 228 | 177 | 300 | 66.0 | 11.0 | 15.0 | 8.0 |
| | | | | Throwaway Prefilters | | | | |
| 1,000 | 17 | 44.50 | 21.30 | 99.30 | 11.4 | 18.0 | 24.7 | 45.9 |
| 10,000 | 10 | 25.00 | 17.05 | 36.15 | 10.9 | 29.7 | 24.7 | 34.7 |

[a]From First and Silverman, ref. 28.

primarily in the degree of detail and level of effort required for each step. These principal steps, as they would apply to a complete facility, such as a power plant, or to a portion of a facility are discussed below.

1. Obtain a clear interpretation of the facility for which the estimate is to be prepared. Sufficient technical information must be available to thoroughly describe the facility. This may include preliminary drawings and specifications and other information that will enable the estimator to visualize the entire project from its conception to its completion. When the item of particular interest is only a part of a larger installation, as in the case of containment vessels, and the estimate is for the purpose of determining the most economical of several alternate designs, it is still necessary to consider the entire installation for each design, since changes in one part of the plant may significantly affect the cost of other parts of the plant. For even the most preliminary estimate, a conceptual drawing indicating the types and quantities of materials and the methods of construction is required. Detailed estimates may require a rather detailed preliminary design of the entire installation.

2. Define the scope of the estimate. It is particularly important for detailed estimates that there be a clear understanding of the items and scope of the services to be included in the estimate. The scope of responsibilities will vary from one project to another and the extent to which these items are included will significantly affect the cost. Responsibilities that may or may not be included in the final estimate, but which still must be considered, include the following: guarantees, interest during construction, insurance premiums, taxes, fees, cost of land, cost of training operators, services to be provided for the client (such as assisting with the preparation of applications for licenses or permits), costs of consultants (if required), cost of utilities (such as water, electric power, and waste disposal), and access roads or railroads. Many other items fall in this same category.

3. Collect and analyze reference materials. All available material concerning the terms and conditions of the project and any other information affecting the manner in which the project will be carried out should be carefully examined. Available historical information on experiences with comparable types of projects and any previous experience with the same client should be reviewed.

4. Develop basic policies and evaluation factors for the estimate. The basis of the estimate and the factors to be included should be carefully spelled out in order to clarify the estimate and to avoid misunderstandings in the use of the cost figures obtained. This is done for the purpose of defining the estimate in the areas not covered in the specifications and other documents or in areas where the available information is not completely clear. The inclusions, exclusions, options, and assumptions on which the estimate is based must be specified. This step is, in effect, a consolidation of all the information developed in steps 1, 2, and 3.

5. Determine equipment and material availability. A survey should be made of the general market conditions for the kinds of equipment and materials required for the facility to determine the current availability and lead time from order to delivery of the major items.

6. Analyze labor market. All conditions affecting the cost of labor at the project location should be investigated. The information required includes labor availability and wages, labor productivity, local labor practices, and other labor related expenses, such as travel, subsistence, and paid costs of fringe benefits. Productivity factors may be established in order to relate labor costs at the project location with labor costs in other areas where more specific experience is available.

7. Evaluate site conditions. In addition to the general availability and costs of labor and materials as affected by local conditions, many other local factors must be evaluated to determine their effect on the overall cost of the project. Climatic conditions must be studied along with the project schedule in order to determine what provisions, if any, are necessary for winter protection, snow removal, air-conditioning, etc., for both the installation to be constructed and for the temporary construction facilities. The availability of utilities such as water and electric power required for construction, the availability and condition of access roads and railroads, and the avilability of space for temporary construction facilities should all be determined. Local building codes and other local requirements such as sales taxes, use tax, and other taxes should be reviewed to determine their effect on the construction cost.

8. Determine quantities of material and equipment. The quantities of all types of materials and equipment required for the project should be determined by a careful examination of the drawings and specifications and other technical information related to the project. The items should be grouped in units to facilitate pricing. Emphasis should be placed on the items that will result in large blocks of cost because of the large quantities involved or the high cost of certain items.

9. Pricing. Prices should be assigned to the materials and equipment, labor, services, and all other items affecting the total cost of the project. Items to be priced may include technical services, procurement services, preliminary operations, testing, and plant startup. Pricing of all items should take into consideration the project schedule and the anticipated working conditions. Pricing of materials and equipment should be based on a thorough evaluation of current market conditions. It may require consultation with manufacturers and technical experts, and it should include evaluation of bids from suppliers. Pricing of other services to be provided should be based almost entirely on an analysis of past experiences related to the conditions of the project.

10. Determine contingencies. Each item in the estimate should be analyzed to determine the validity of the information used in order to establish the degree of confidence placed on the estimated cost of the item. The analysis should include a determination of the firmness of the design, as well as the basis of pricing. This analysis will determine the potential saving or increases in cost. By applying proper judgment, the risks involved and, in turn, the contingency required should be established. It is in this area of judgment that the past experience and capability of the estimator and a thorough understanding of all factors involved in the project are essential. There are no rules or set patterns to provide guidance, and it is frequently the soundness of this judgment that will make the difference between a good and a poor estimate.

11. Review final estimate. The final estimate should be reviewed to determine that it conforms with the technical data and project description and is consistent with the policies established for making the estimate.

### 11.6.3 Detailed Cost Estimate of the HWCTR Containment Structure

To illustrate the manner in which a detailed cost estimate may be prepared, an estimate of the cost of the containment vessel for the HWCTR is presented in Table 11.20. These costs correspond with the figures presented for the HWCTR in Table 11.10, but they are presented in considerably more detail. The extent and form of the cost information presented in Table 11.20 are representative of a cost summary that might be developed for budgetary purposes. The HWCTR containment vessel is not typical of the containment vessels used for most power reactors (see Chap. 7) and the costs used cannot be considered representative of other locations, but the type of information provided is typical.

This estimate covers the excavation, substructure, and superstructure of the composite concrete and steel containment vessel. The vessel is a prestressed concrete cylinder, 70 ft in diameter, topped by a hemispherical steel dome. The steel dome contains the air locks and other penetrations. The structure rests on a reinforced concrete mat 57 ft below grade. The plant is located at the Savannah River Plant of the AEC at Aiken, South Carolina.

The methods used to estimate installed costs of large structures such as this vary widely between organizations. The following tabulation represents one way in which cost accounts were maintained and presented, but other accounting systems may be more suited to the operations of other organizations. The accounting system presented in the AEC cost evaluation guide,[1] represented by Account 219 in the Appendix, is a useful standard system to permit comparisons of costs on similar projects, but contractors have not often used the system if it differs from their normal practice. Reassigning costs to a system different from that for which they were obtained is quite arbitrary, as indicated in Section 11.4.

## 11.7 SUMMARY

In this chapter a number of the principal cost items of reactor containment are identified and discussed. Published cost information on constructed containment systems is presented, and various economic studies of containment are reviewed. In attempting to develop general and useful containment cost data from information such as this, several problems arise. First, it is difficult to separate cost components that are principally associated with containment from those that are also associated with the conventional plant systems and enclosures. Second, even if the containment cost components can be properly designated, the data, whether from actual construction cost experience of from careful estimates, will be strictly applicable only to a specific design concept at a specific

Table 11.20. HWCTR Containment Vessel Construction Cost Estimate[a]

| Item | Quantity | Cost ($) Labor | Material | Total |
|---|---|---|---|---|
| Excavation and backfill | | | | |
| Excavation, including allowance for for pumping, slope protection | 70,000 $yd^3$ | 13,200 | 2,660 | 15,860 |
| Backfill | | | | |
| Compacted | 38,000 $yd^3$ | 19,000 | | 19,000 |
| Standard | 37,000 $yd^3$ | 8,000 | | 8,000 |
| Borrow | 5,000 $yd^3$ | 1,000 | | 1,000 |
| Concrete | | | | |
| Work slab | 200 $yd^3$ | 2,100 | 3,200 | 5,300 |
| Foundation mat | 845 $yd^3$ | 9,100 | 13,500 | 22,600 |
| Exterior walls | 610 $yd^3$ | 13,000 | 19,000 | 32,000 |
| Interior walls | 805 $yd^3$ | 4,700 | 13,300 | 18,000 |
| Floor slabs (except 0) | 216 $yd^3$ | 1,600 | 1,900 | 3,500 |
| Shielding concrete | 386 $yd^3$ | 4,600 | 3,400 | 8,000 |
| Barytes concrete | 150 $yd^3$ | 1,500 | 7,200 | 8,700 |
| Pneumatic mortar | 100 $yd^3$ | 6,000 | 2,000 | 8,000 |
| 0 floor slab (including 6-in. topping) | 825 $yd^3$ | 6,600 | 10,800 | 17,400 |
| Forms | | | | |
| Foundation mat | 1,250 $ft^2$ | 1,400 | 500 | 1,900 |
| Exterior walls | 19,900 $ft^2$ | 62,400 | 8,000 | 70,400 |
| Interior walls | 16,500 $ft^2$ | 25,500 | 6,600 | 32,100 |
| Shielding | 5,700 $ft^2$ | 8,800 | 7,300 | 11,100 |
| Floor slabs (including 0) | 8,850 $ft^2$ | 24,300 | 7,600 | 31,900 |
| Reinforcing steel | | | | |
| Foundation mat | 126,500 lb | 3,200 | 10,500 | 13,700 |
| Exterior walls | 251,000 lb | 9,600 | 20,000 | 29,600 |
| Interior walls | 110,950 lb | 11,300 | 9,200 | 20,500 |
| Floor slabs (except 0) | 20,200 lb | 2,250 | 1,700 | 3,950 |
| Working slab, mesh | 6,750 lb | 1,000 | 1,100 | 2,100 |
| 0 floor slab | 90,600 lb | 3,000 | 7,500 | 10,500 |
| Structural steel | | | | |
| Steel containment shell, including | | | | |
| Steel plate reservoir<br>Personnel air lock<br>Emergency air lock<br>Access door<br>Cut and close temporary opening<br>Prime paint interior surface<br>Prime paint interior surface of reservoir<br>Prestressing of anchor bolts<br>Radiograph inspection<br>Anchor bolt template<br>Welding seal ring | | 120,000 | 200,000 | 320,050 |
| Allowance for additional hanger supports | | | 2,000 | 2,000 |
| Crane runway and support<br>Stainless steel floor framing<br>Stainless steel stair framing<br>Stainless steel platform framing | 75 tons | | 27,500 | 27,500 |

[a]From ref. 30.

Table 11.20 (continued)

| Item | Quantity | Cost ($) | | |
|---|---|---|---|---|
| | | Labor | Material | Total |
| Additional steel for hangers | | 500 | 2,000 | 2,500 |
| Field erection of structural steel | 75 tons | 7,500 | 500 | 8,000 |
| Stair treads | 75 each | 1,000 | 1,000 | 2,000 |
| Steel floor grating | 5,300 $ft^2$ | 2,300 | 9,000 | 11,300 |
| Steel ladders | 50 lin ft | 1,000 | 1,000 | 2,000 |
| Handrails | 475 lin ft | 1,500 | 600 | 2,100 |
| Miscellaneous steel and anchor bolts | | 2,000 | 1,000 | 3,000 |
| Doors | | | | |
| 4- × 7-ft shielding doors | 5 each | 2,000 | 8,500 | 10,500 |
| Door operators and trolleys | 5 each | 1,000 | 4,960 | 5,960 |
| Semirevolving doors | 2 each | 300 | 500 | 800 |
| Special single swing doors | 3 each | 300 | 300 | 600 |
| Q deck stair enclosure | | 1,500 | 2,300 | 3,800 |
| Counterweighted door sheaves | 4 each | 800 | 2,000 | 2,800 |
| Painting | | | | |
| External surface of containment shell | 16,000 $ft^2$ | 1,000 | 600 | 1,600 |
| Structural steel | | 2,000 | 1,000 | 3,000 |
| Steel doors and frames | | 100 | 50 | 150 |
| Miscellaneous steel | | 1,000 | 500 | 1,500 |
| Interior surface of containment shell | 16,000 $ft^2$ | 2,400 | 1,200 | 3,600 |
| Surface of 15,000-gal water tank | 2,000 $ft^2$ | 800 | 400 | 1,200 |
| Interior concrete surfaces | 50,000 $ft^2$ | 5,000 | 3,000 | 8,000 |
| Lighting | | | | |
| Incandescent units | 116 each | 4,970 | 2,750 | 7,720 |
| Shower lights | 2 each | 80 | 40 | 120 |
| Special fixtures and receptacles | 4 each | 600 | 800 | 1,400 |
| Emergency units | 14 each | 700 | 1,540 | 2,240 |
| Normal lighting panel, transformer, and feeder | 1 each | 2,400 | 2,000 | 4,400 |
| Emergency lighting panel and feeder | 1 each | 1,600 | 900 | 2,500 |
| Grounding | | 1,200 | 600 | 1,800 |
| Plumbing | | | | |
| Floor drains and associated lines | | | | |
| At 0 elevation, six 4-in. floor drains<br>At 37 ft 6 in. elevation, three 4-in. floor drains<br>At 52 ft 6 in. elevation, eight 4-in. floor drains | | 3,400 | 1,600 | 5,000 |
| Safety showers | 2 each | 1,200 | 900 | 2,100 |
| Hose bibs | 4 each | 100 | 200 | 300 |
| Hood sampling trough | | 50 | 200 | 250 |
| Insulated water lines | | 350 | 150 | 500 |
| Miscellaneous | | | | |
| Prestressing strands and fittings | | | 30,000 | 30,000 |
| Installation labor | | 5,000 | | 5,000 |
| External steel pilasters | 28 tons | 6,250 | 5,800 | 12,050 |
| High-strength bolts | 335 each | | 1,835 | 1,835 |
| Cable-stressing equipment, including | | | | |
| 130-ton pneumatic center-hole jacks<br>Jacking chairs<br>Calibrated gages<br>Hose and quick couplers<br>Pneumatic pump and accessories | 9 each<br>8 each<br>8 each | 1,000 | 4,800 | 5,800 |
| Scaffold for prestressing and pneumatic mortar | | 5,000 | | 5,000 |

Table 11.20 (continued)

| Item | Quantity | Cost ($) | | |
|---|---|---|---|---|
| | | Labor | Material | Total |
| Concrete floor finish | 10,000 $ft^2$ | 1,400 | 500 | 1,900 |
| Precast concrete floor plugs | 10 each | 15,000 | 5,000 | 20,000 |
| Precast slab and beam | | 2,160 | 1,010 | 3,170 |
| 80-lb rail for coffin carriage, including shimming and alignment | | 2,000 | 1,500 | 3,500 |
| Steel containment shell insulation | 16,000 $ft^2$ | 12,000 | 24,000 | 36,000 |
| Placing anchor bolts, seal ring, water stop, etc., for containment shell | | 10,100 | 4,300 | 14,400 |
| Truscon inserts | 12,000 lin ft | 600 | 1,800 | 2,400 |
| Waterproof exterior of concrete wall | 11,000 $ft^2$ | 660 | 440 | 1,100 |
| Allowance for vacuum test and mockup of core wall | | 5,000 | 3,000 | 8,000 |
| Three coat "Liquid Tile" application on interior surface of concrete shield | | 5,000 | 3,000 | 8,000 |
| Allowance for pressure and leakage rate test | | 15,000 | 5,000 | 20,000 |
| Subtotal | | 501,170 | 525,185 | 1,026,355 |
| Manpower and materials distribution expense, processing, and security requirements | | 5,830 | | 5,830 |
| Allowance for design modifications | | 20,000 | 2,815 | 22,815 |
| Subtotal | | 527,000 | 528,000 | 1,055,000 |
| Premium time | | 4,000 | | 4,000 |
| Allowance for advancing wages and material prices | | 7,000 | 2,000 | 9,000 |
| Maintenance of construction facilities, safety meetings, reporting time, etc. | | 47,000 | 23,000 | 70,000 |
| Subcontractors' fees | | | 7,000 | 7,000 |
| Net total | | 585,000 | 560,000 | 1,145,000 |

location and under specific market conditions. Third, if costs for complete containment systems that have been constructed are presented, the effects of escalation, experience gained through the construction and test phases, and advancements in technology are difficult to evaluate. Finally, costs for complete systems may include hidden items for design features that are considered neither necessary nor desirable for another installation. Considering these problems, the conclusions from the various economic studies discussed in this chapter must be viewed carefully in total with the ground rules and the assumptions that were necessary for the study.

The preferred method for cost estimation is to base all estimates on a relatively well-developed containment design and to take into consideration the actual quantities of materials and equipment involved at a real site with current prices. In cases where this is not possible, methods that approach this to the maximum possible degree should be used. Total system estimates made by others for different conditions or for another site should only be used as a check to uncover possible errors or omissions.

Many of the cost studies and total containment cost estimates now available may be of limited value for future applications because of the very rapid progress being made in the application of nuclear power plants to large central stations. It is now quite obvious that the trend toward very large nuclear units [500 to 1000 Mw(e)] will continue with an accompanying improvement in economics due both to the larger sizes and to improvement in design. On the other hand, siting problems will become more severe and the cost of containment and other safeguard features may increase substantially. Although the containment features that will be required in future plants cannot be predicted, and although both the total cost of containment and the relationships between costs of various containment components may change substantially, the containment cost experience reflected in this chapter will form the basis for selecting future containment system designs and for estimating the costs of these systems.

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# 12. RESEARCH

J. R. Buchanan

## 12.1 INTRODUCTION

The fundamental purpose of reactor containment is to protect the public from nuclear hazards, even in the unlikely event of a reactor accident. In order to do this, it is necessary to understand the characteristics of the various credible accidents for which some containment is necessary and to define the fission-product-retaining capabilities of the various containment barriers available in different types of systems. As examples of the wide-range of information needed to permit adequate containment designs, the following can be mentioned: structural loads due to coolant decompression and consequent response of complicated structures; transient and long term heat transfer; thermodynamic and hydrodynamic mechanisms influencing magnitudes and duration of loadings; effects of high temperature, high pressure, random and complex mechanical loads, and irradiation on primary system integrity; characteristics of brittle and ductile fracture and theories of fracture mechanics; field erection and construction practices; energy release in metal-water reactions; chemical and physical form of activity released and its transport mechanisms; the leak rate phenomena of large structures; and the reliability, efficiency, and mode of operation of engineering safeguards, such as filters or sprays.

Much of the research devoted to gathering data on such topics is carried out under the auspices of the AEC through their Assistant Director for Nuclear Safety in the Division of Reactor Development. The program covers topics such as the structural response of shells and the integrity of primary vessels and piping, sources of loading (mechanical and thermal) on containment systems, such as the loss-of-coolant process itself, the release of fission products, the natural transport properties and countermeasures that can be taken to reduce transport, pressure-suppressing characteristics of pools and sprays, and advanced containment systems. Analysis and small- and large-scale testing are all employed. In addition to the research administered by the AEC's Nuclear Safety group, there are independent studies under way as a part of specific reactor programs.

In the following sections, brief descriptions of the major containment research projects are given along with summaries of references that describe the work. Both research programs presently under way (1964) and those recently completed are cited. No attempt has been made to include investigations under way in other countries.

## 12.2 STUDIES OF THE RESPONSE OF VESSELS, PIPING, AND OUTER CONTAINMENT STRUCTURES TO ACCIDENTS

Studies reviewed in this section are concerned with the response of the primary system and of the outer containment vessel to the loads that could be imposed on them in credible reactor accidents. The work has been devoted, in general, to defining (1) the mechanics of possible primary system failure, (2) the ability of containment systems to resist transient and long-term loads without significant loss of integrity, (3) the ultimate resistance of containment systems to rapid pressure loading or missile formation, and (4) means of improving current containment design. Programs with these purposes are described in Table 12.1 and reviewed below.

### 12.2.1 Aberdeen Proving Ground

The research program at Aberdeen was recently completed. In this work, an attempt was made to define the upper limit response of simple containment shells to large transient loads and the vibrational modes of thin shells under transient internal pressure in the elastic and plastic ranges. Mathematical formulations for spheres, shallow spherical segments, and capped cylinders were developed and verified by tests in the elastic range. Empirical tests were conducted to determine the upper limit internal pressure and dilations for certain cylindrical and spherical shells in sizes up to diameters of 20 ft. The scaling of elastic vibrations and of some gross plastic deformations was examined by model tests of shells and cantilever beams.

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2. W. E. Baker and F. J. Allen, The Response of Elastic Spherical Shells to Spherically Symmetric Internal Blast Loading, Report BRL-M-1113, Ballistic Research Laboratories, November 1957.

3. W. E. Baker, W. O. Ewing, Jr., and J. W. Hanna, Laws for Large Elastic Response and Permanent Deformation of Model Structures Subjected to Blast Loading, Report BRL-1060, Ballistic Research Laboratories, December 1958.

4. J. W. Hanna, W. O. Ewing, Jr., and W. E. Baker, The Elastic Response to Internal Blast Loading of Models of Outer Containment Structures for Nuclear Reactors, Report BRL-1067, Ballistic Research Laboratories, February 1959.

5. W. E. Baker, The Elastic-Plastic Response of Thin Spherical Shells to Internal Blast Loading, Report BRL-M-1194, Ballistic Research Laboratories, February 1959.

Table 12.1. Studies of the Response of Vessels, Piping, and Outer Containment Structures to Accidents

| Investigator | Person in Charge | Programs | Method of Reporting |
|---|---|---|---|
| Aberdeen Proving Grounds | O. T. Johnson | Vibration and dilation of shells under transient internal pressures | Bimonthly progress reports and topical reports |
| Illinois Institute of Technology Research Institute | E. V. Gallagher | Studies of dilation of shells, effects of penetration, and analysis of shock transmission | Bimonthly, topical, and annual reports |
| Atomics International | Arvin Gibson | Reactor housing and leak testing of conventional building components | Quarterly and annual reports |
| Battelle Memorial Institute | T. J. Atterbury | Analysis of theoretical and experimental data for reinforcement of openings in pressure vessels | Bimonthly progress and topical reports |
| General Electric Company | E. R. Kilsby | Pipe rupture literature survey | Monthly progress and topical reports |
| Naval Ordnance Laboratory | W. R. Wise, Jr. | Response of model cylinders to internal blast | Bimonthly progress and topical reports |
| Southwest Research Institute | A. G. Pickett | Cyclic pressure tests of large-size pressure vessels and evaluation of weld defects in fatigue | Bimonthly progress reports, phase reports, and final reports |
| Stanford Research Institute | G. R. Fowles | Generation of missiles and penetrability of containment | Monthly, annual, and final reports |

6. W. C. Olson and Harry Goldstein, Air Blast Measurements Around Water-Filled Simulated Nuclear Reactor Core Vessels, Report BRL-M-1219, Ballistic Research Laboratories, July 1959.

7. W. H. Hoppmann, II, and C. E. Miller, Forced Vibrations of Elastic Orthotropic Shells, Rensselaer Polytechnic Institute, May 1960.

8. W. E. Baker, The Axisymmetric Modes of Vibration of Spherical Shell, Report BRL-1122, Ballistic Research Laboratories, December 1960.

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10. J. W. Hanna and W. O. Ewing, Jr., Effectiveness of Lining Materials in Increasing the Blast Resistance of a Simulated Outer Containment Vessel for a Nuclear Reactor, Report BRL-M-1341, Ballistic Research Laboratories, April 1961.

11. R. C. Makino and J. Giese, Underwater Expansion of a Blast Loaded Thin Spherical Shell, Report BRL-1153, Ballistic Research Laboratories, October 1961.

12. W. H. Hoppmann, II, Flexural Vibrations of Shallow Orthotropic Spherical Shells, Rensselaer Polytechnic Institute, January 1962.

13. W. E. Baker, J. W. Hanna, and B. J. Lindenberger, Elastic Response to Blast Loading of Strain-Rate Dependent Cantilevers, Report BRL-1179, Ballistic Research Laboratories, November 1962.

14. W. H. Hoppmann, II, and C. N. Baronet, A Study of the Vibrations of Shallow Spherical Shells, Report BRL-1188, Ballistic Research Laboratories, January 1963.

15. I. A. Minkarah and W. H. Hoppmann, II, Flexure of Circularly Anisotropic Circular Plate with Eccentric Load, Rensselaer Polytechnic Institute, February 1963.

16. I. A. Minkarah and W. H. Hoppmann, II, Flexural Vibrations of Cylindrically Orthotropic Circular Plates, Rensselaer Polytechnic Institute, October 1963.

17. W. E. Baker, The Elastic Response of Thin Spherical Shells to Internal Blast from Eccentrically Placed Explosive Charges, Report BRL-M-1520, Ballistic Research Laboratories, November 1963.

18. W. H. Hoppmann, II, M. I. Cohen, and V. X. Kunukasseril, Elastic Vibrations of Paraboloidal Shells of Revolution, Rensselaer Polytechnic Institute.

19. W. E. Baker, J. W. Hanna, and J. D. Patterson, Response of Submerged Spherical Shells to Internal Blast Loading, Report BRL-1231, Ballistic Research Laboratories, January 1964.

20. J. W. Hanna and W. O. Ewing, Jr., The Plastic Response to Internal Blast Loading of Models of Outer Containment Structures for Nuclear Reactors, Report BRL-M-1530, Ballistic Research Laboratories, January 1964.

### 12.2.2 Illinois Institute of Technology Research Institute

The Institute is developing methods for predicting the semistatic and dynamic dilations of thin shells in the plastic range under internal pressure. Tests with model cylinders are being used to verify the semistatic predictions. The weakening effects of penetration on shell response and the reinforcement required to restore shell strength are also under study.

The Institute is also engaged in the analysis of blast shields for modifying the pressure loads experienced by vessels or shells. The properties of crushable shield materials are being determined by small-scale tests under shock loading.

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2. P. Lieberman, Shock Impingement Experiments of Crushable Solids, USAEC Report ARF-4132-10, Armour Research Foundation, June 1959.

3. Armour Research Foundation, Studies of Reactor Containment, Summary Report No. 1, February 1, 1959, to July 31, 1959, USAEC Report ARF-4132-11, Aug. 1, 1959.

4. T. A. Zaker, Point Source Explosion in a Solid, USAEC Report ARF-4132-6, Armour Research Foundation, November 1959.

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6. E. A. Brown, Experiments on the Explosive Decompression of Water, USAEC ARF-4132-9, Armour Research Foundation, November 1959.

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12. P. Lieberman, Studies of Reactor Containment, Bimonthly Progress Report November-December 1962, USAEC Report ARF-K-6018-4, January 1962.

13. Approximate Bursting Strength of Thin-Loaded Cylinders with Hemispherical Caps, USAEC Report ARF-4132-17, Armour Research Foundation, May 1962.

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15. N. A. Weil, Studies of Reactor Containment, Bimonthly Progress Report November-December 1962, USAEC Report ARF-8262-4, Armour Research Foundation, Jan. 21, 1963.

16. E. V. Gallagher, Studies of Reactor Containment, Bimonthly Progress Report April-May 1963, USAEC Report IITRI-A-6042-1, Illinois Institute of Technical Research Institute.

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21. M. Salmon, Plastic Instability of Cylindrical Shells with Rigid End Closures, USAEC Report ARF-4132-19, Armour Research Foundation, 1964.

22. P. Lieberman, Plastic Stress Wave Attenuation in a Porous Solid (to be issued).

### 12.2.3 Atomics International

A program was completed at Atomics International for defining the leakage of air that can be expected through components of conventional buildings used to house reactors. Metal and concrete panels, various types of construction joints and concrete mixes, typical penetrations, and various paints, resins, and sealants were tested. Small and highly simplified model buildings were used to attempt to verify the validity of summing the leakage of individual components to predict the leakage of a large building. Leakage coefficients were determined, investigated for reproducibility, and tabulated for various building components. A design manual for conventional buildings of this type was written (see report NAA-SR-10100).

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4. R. L. Koontz et al., Conventional Building for Reactor Containment, USAEC Report NAA-SR-10100, North American Aviation (to be published).

5. C. T. Nelson, A. S. Gibson, and R. L. Koontz, Prediction and Control of Leakage Through Cracks and Capillaries in Reinforced Concrete Buildings Used to House Reactors, USAEC Report NAA-Sr-Memo-9429, North American Aviation, March 6, 1964.

### 12.2.4 Battelle Memorial Institute

A reasonable set of rules covering reinforcement of openings in pressure vessels is needed for incorporation into the ASME Pressure Vessel Code and the ASA Piping Code. A detailed analysis and comparison of the available theoretical and experimental reinforcement data is being made at BMI with this need in mind. The study will result in formulas and curves that will permit the quantitative evaluation of stresses in commonly used nozzle and piping arrangements. This information is necessary for fatigue or other detailed stress analysis. The primary source of input is the data developed by the Pressure Vessel Research Committee of the Welding Research Council of The Engineering Foundation over the past eight years. References: None as yet.

### 12.2.5 General Electric Company

Literature surveys have been completed at the General Electric Company, San Jose, on pipe-rupture histories, both foreign and domestic. The surveys include development of information on fracture modes and mechanics. Industrial pipe designers and stress analysts are being surveyed concerning important methods and problems now existing in piping design. The reports listed below summarized what is now known or can be predicted about pipe failures. One report recommends an experimental program for future work.

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3. General Electric Company, Survey of Water Reactor Primary System Conditions Pertinent to the Study of Pipe Rupture, USAEC Report GEAP-4445, January 1964.

4. General Electric Company, Survey of Piping Failures for the Reactor Primary Coolant Pipe Rupture Study, USAEC Report GEAP-4574, May 1964.

5. General Electric Company, Experimental and Analytical Program Recommendations Reactor Pipe Rupture Study, USAEC Report GEAP-4474, January 1965.

### 12.2.6 Naval Ordnance Laboratory

A program for defining the maximum resistance of model thick-walled vessels to internal transient pressures produced by explosive charges has been completed at the Naval Ordnance Laboratory. The program included studies of (1) the effect of ductility, hardness, ultimate strength, and strain rate on the ability of reactor vessels to withstand a simulated nuclear excursion, (2) the effect of the rate of energy release by propellant and explosive energy sources, (3) the effect of geometry, wall thickness, and scaling factor, and (4) the partition of energy released by the charge into mechanical and thermal effects.

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5. W. R. Wise, Jr., and J. F. Proctor, Explosion Containment Laws for Nuclear Reactor Vessels, USAEC Report NOL-TR-63-140, Naval Ordnance Laboratory (to be published).

6. W. R. Wise, Jr., J. F. Proctor, and L. P. Walker, Response of Enrico Fermi Reactor to TNT-Simulated Nuclear Accidents, USAEC Report NOL-TR-62-207, Naval Ordnance Laboratory, November 1964.

### 12.2.7 Southwest Research Institute

Studies for determining the low-cycle fatigue strength and mode of terminal failure of full-sized pressure vessels are in progress. The effects of design geometry, materials properties, fabrication procedures, and quality-control inspection standards are being investigated by comparing the data from conventional laboratory-specimen tests with the results from tests of pressure vessels of different sizes. To date, seven full-sized vessels of three different materials have been tested to failure. Tests are in progress on vessels one-half full size and full-sized vessels with different kinds of strain intensifiers than were studied in the original set. A tentative design analysis procedure for predicting vessel life was demonstrated to be conservative.

Another program at the Institute is for studying the effects of fabrication and quality-control inspection procedures on materials properties. Specifically, an investigation is being made of the effects on fatigue life of code-allowable weld and cladding defects as a function of type, size, location, and orientation and the effects of fabrication procedures on the nil ductility temperature. The studies include efforts to correlate nondestructive inspection procedures with consequent accepted defects and reduction in fatigue life and to associate fabrication processes with changes in fracture toughness from base plate to finished vessel or piping.

References

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### 12.2.8 Stanford Research Institute

The Stanford program, which was recently concluded, was designed for determining the type, number, and size of missiles that could be generated as a result of a violent nuclear excursion and for establishing the penetrability of containment shells. Simplified models (1/24, 1/12, and 1/6 scale) of primary vessels and adjacent shielding were subjected to programmed energy releases simulating scaled excursions. Data were obtained for the average size of missiles and the probable distributions around the average. A formula was developed for determination of the energy required for steel and concrete missiles of various geometries to penetrate steel plates of varying thickness and material properties.

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## 12.3 LOSS-OF-COOLANT ACCIDENT STUDIES

One of the most serious accidents that can conceivably occur in power reactor systems of the pressurized- or boiling-water type is loss of coolant. The parameters affecting the behavior and consequences of such accidents are being investigated in laboratory-scale experiments and will soon be studied in large-scale engineering field tests. The latter tests will simulate those of actual power reactor operation as closely as possible. The various loss-of-coolant accident studies are described in Table 12.2.

### 12.3.1 Illinois Institute of Technology Research Institute

The Institute is examining the explosive decompression of water in an attempt to determine the maximum pressure loads on containment structures resulting from a loss-of-coolant accident. Boiling- and pressurized-water conditions are simulated in small-scale vessels. The generation of air shocks, decompression mechanisms in the pressurized medium, jet impact forces, equilibrium pressures, and the existence of metastable flow processes are being investigated. The studies will include both one-dimensional and three-dimensional receivers for the escaping fluids so as to simulate realistic accident conditions and to aid in extrapolating model results to large systems. Structural loadings on reactor internal components due to blowdown will also be measured in small-scale model tests.

#### References

See references in Section 12.2.2.

### 12.3.2 Babcock & Wilcox

Studies of the blowdown of simulated reactor systems directly into pools of water are being conducted by Babcock & Wilcox. The effectiveness of vapor suppression is being examined as a function of the size of the blowdown and the depth of the jet submergence. The various loading

Table 12.2. Loss-of-Coolant Accident Studies

| Investigator | Person in Charge | Programs | Method of Reporting |
|---|---|---|---|
| Illinois Institute of Technology Research Institute | E. V. Gallegher | Explosive decompression of water | Bimonthly, topical, and annual reports |
| Babcock & Wilcox | | Vapor suppression studies | |
| Hanford Laboratories | E. R. Irish | Containment system experiment (CSE) | Quarterly and topical reports |
| Phillips Petroleum Co. | T. R. Wilson | Loss of flow tests (LOFT) | Topical and quarterly technical reports and semiannual program review |
| University of Minnesota | H. Isbin | Studies of two-phase flow in pipes and through orifices | Bimonthly, annual, and final reports |

forces (shock, impact, transient, and static) that would follow a system rupture are also being investigated. Rupture tests of vessels up to 6 in. in diameter have been conducted. Initial tests were performed in an open tank. The program just completed utilized a closed tank, and the degree of condensation was determined by the pressure in the tank. Babcock & Wilcox are also subcontractors to Phillips Petroleum on the LOFT program and are responsible for conducting analytical studies of anticipated fluid behavior during a transient.

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2. Babcock & Wilcox, Closed Tank Vapor Suppression Program, USAEC Report BAW-3084-1, 1964.
3. Babcock & Wilcox, Study and Recommendations Relative to Loss of Fluid Test Facility, High Versus Low Pressure Design, USAEC Report BAW-1288, November 1963.

### 12.3.3 Hanford Laboratories

The Containment Systems Experiment (CSE) at Hanford will use a model primary vessel that will discharge into various types of containment (simple shell, pool suppression, spray suppression, etc.) for studying the significance of the loss-of-coolant accident on the efficiency and reliability of the various types of containment. Included in the investigations will be the ability to withstand mechanical loads, reduce pressures, and retain fission-product activity. Both boiling- and pressurized-water conditions are to be studied with respect to size and location of simulated pipe failures. Simulated and real fission products will be used to assess the correlation of activity transport with the magnitude and rate of coolant loss. Separate studies of the blowdown process are also being conducted.

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### 12.3.4 Phillips Petroleum Company

Engineering-scale experiments are to be conducted at NRTS to demonstrate the behavior and consequences of loss-of-coolant accidents in water-cooled reactors. A complete experimental pressurized-water reactor system of about 50-Mw thermal power that is typical of existing pressurized-water power reactor systems will be fabricated and operated for significant periods to achieve a large fission-product inventory and the resulting heat fluxes necessary to perform a meaningful experiment. Prior to operation with fuel, the loss-of-coolant accident will be investigated by rupturing diaphragms of various sizes in the primary coolant system to determine flow rates, decompression effects within the reactor vessel, piping, etc. The subsequent experiment with an irradiated core will be conducted to provide information regarding the extent of core damage, percentage and distribution of fission product released, activity levels, existence of chemical reactions, containment vessel leakage, and fission-product transport.

Posttest examination of the reactor fuel and components will be performed to aid in evaluation of the experiment and correlation with existing experimental data. Parallel studies to develop calculational techniques for predicting accident consequences and establishing accident criteria will be performed.

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3. O. M. Hauge and D. R. Mousseau, Preliminary STEP Facility Concepts on NTF and LOFT, USAEC Report PTR-597, Phillips Petroleum Company, Sept. 4, 1962.

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6. E. E. Erickson (Ed.), Preliminary Site Evaluation Report LOFT Facility, USAEC Report PTR-644, Phillips Petroleum Company, June 14, 1963.

7. T. R. Wilson, O. M. Hauge, J. M. Waage, and G. B. Matheney, Review of Loss-of-Coolant Safety Tests and LOFT Facility, USAEC Report PTR-645, Phillips Petroleum Company, June 14, 1963.

### 12.3.5 University of Minnesota

The aim of the program at the University of Minnesota is to define the maximum rate of flow for orifices in walls of a small model primary vessel by test and by mathematical analysis. The group at Minnesota has studied various sizes of orifices (all small in relation to the size of the vessel), pressure profiles near orifices, and mass flow rates. In addition to work on transient blowdown, a correlation has been developed for critical two-phase steady flow in pipes. Linearized mathematical models of pressure suppression systems that show the significance of flow rates, size of receiver, size of vent pipe, etc., during the loss-of-coolant accident have also been developed.

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## 12.4 METAL-WATER REACTIONS AND METAL IGNITIONS

In a reactor accident, the reaction of the reactor coolant with fuel or structural material can result in a significant contribution to the energy released. If the reaction rate is rapid, high pressures may be generated that result in excessive damage to the system. Laboratory and in-pile experiments on metal-water reactions and metal ignitions are being conducted to study the problem. With increased emphasis on the development of fast and thermal breeder systems, studies of the effects of liquid metal and gaseous coolants on fuel and cladding materials will be initiated. The studies now under way are described in Table 12.3 and discussed below.

### 12.4.1 Argonne National Laboratory

Metal-water reactions are under study in both in-pile and out-of-pile experiments at the Argonne National Laboratory. The metals being investigated include aluminum, zirconium, stainless steel, and uranium. Test methods include rapid heating of metal wires in a water environment, steam pressure pulse techniques, and autoclave excursion tests in the TREAT facility. The principal method of investigation so far has been the heated wire technique. Equations are being developed to describe the reaction rate of the various metals with water or steam as a function of temperature,

Table 12.3. Metal-Water Reactions and Metal Ignitions

| Investigator | Person in Charge | Programs | Method of Reporting |
|---|---|---|---|
| Argonne National Laboratory | L. Baker, Jr. | Metal-water reaction studies and metal oxidation and ignition studies | Monthly, quarterly, and topical reports |
| Atomics International | A. A. Jarrett | Sodium oxidation and fission-product retention | Quarterly and annual reports |

surface area, previous oxidation, etc. Isothermal rate laws are also being determined that can be extended to systems in which the temperature is rapidly changing and where physical limitations of reaction rate may occur. These limitations are being investigated. Calculational procedures will be devised based on the above information that can be used to predict the degree of hazard involved in a potential reactor accident.

Oxidation, ignition, and combustion of uranium, zirconium, plutonium, thorium, and carbides are also being studied to show the effects on pyrophoricity of temperature, pressure, metal purity, metallurgical history, gas composition, type of oxide film, thermal conductivity, degree of subdivision, temperature gradient, and mass. The experiments include isothermal oxidation at various temperatures, ignition temperature determination, and burning rate studies.

References

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2. H. A. Porte, J. G. Schnizlein, R. C. Vogel, and D. F. Fischer, Oxidation of Zirconium and Zirconium Alloys, USAEC Report ANL-6046, Argonne National Laboratory, September 1959.

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6. R. E. Wilson and P. Martin, Metal-Water Reactions: Levitation Method, Chemical Engineering Division Summary Report, April, May, June, 1961, p. 208, USAEC Report ANL-6379, Argonne National Laboratory, 1961.

7. R. C. Liimatainen, R. O. Ivins, M. F. Deerwester, and F. J. Testa, Studies of Metal-Water Reactions at High Temperatures. II. TREAT Experiments Status Report on Results with Aluminum, Stainless Steel-304, Uranium, and Zircaloy-2, USAEC Report ANL-6250, Argonne National Laboratory, January 1962.

8. L. Baker, Jr. and L. C. Just, Studies of Metal-Water Reactions at High Temperatures. III. Experimental and Theoretical Studies of the Zirconium-Water Reaction, USAEC Report ANL-6548, Argonne National Laboratory, May 1962.

9. R. O. Ivins, F. J. Testa, and P. Krause, Metal-Water Reaction Studies in TREAT, Chemical Engineering Division Summary Report, July, August, September, 1962, p. 191, USAEC Report ANL-6596, Argonne National Laboratory, 1962.

10. R. E. Wilson and P. Martin, Metal-Water Reactions: Isothermal Studies of the Uranium-Steam Reaction by the Volumetric Method, Chemical Engineering Division Summary Report, April, May, June, 1962, p. 148, USAEC Report ANL-6569, Argonne National Laboratory, 1962.

11. R. C. Liimatainen, R. O. Ivins, and F. J. Testa, Metal-Water Reaction Studies in TREAT, Chemical Engineering Division Summary Report, January, February, March, 1962, p. 186, USAEC Report ANL-6543, Argonne National Laboratory, 1962.

12. R. E. Wilson, L. Mishler, and C. Barnes, Metal-Water Reactions: Studies of the Aluminum-Water Reaction by the Levitation-Melting Method, Chemical Engineering Division Summary Report, October, November, December, 1962, p. 196, USAEC Report ANL-6648, Argonne National Laboratory, 1963.

13. R. O. Ivins, F. J. Testa, and P. Krause, Studies of the Aluminum-Water Reaction in TREAT, Chemical Engineering Division Summary Report, October, November, December 1962, p. 201, USAEC Report ANL-6648, Argonne National Laboratory, 1963.

14. R. O. Ivins, F. J. Testa, and P. Krause, Studies of the Aluminum Water Reaction in TREAT, Chemical Engineering Division Summary Report, January, February, March, 1963, p. 179, USAEC Report ANL-6687, Argonne National Laboratory, 1963.

### 12.4.2 Atomics International

Out-of-pile sodium fires are being generated at Atomics International in order to examine the character of the combustion products, the extent of combustion, self-extinguishing characteristics, and retention of iodine by the sodium.

#### Reference

1. R. S. Hart, Distribution of Fission Product Contamination in the SRE, USAEC Report NAA-SR-6890, North American Aviation, March 1962.

## 12.5 STUDIES OF FISSION-PRODUCT RELEASE, AEROSOL AND PARTICULATE GENERATION, AND TRANSPORT OF RADIOACTIVITY

Large fission-product inventories are intrinsically associated with large power reactors. The ability to site reactors in or near large centers of population depends to a great degree upon the ability to understand the movement of fission products in credible accidents and to prevent their release from the containment system. Programs that have this purpose are summarized in Table 12.4 and are discussed in this section.

Table 12.4. Studies of Fission-Product Release, Aerosol and Particulate Generation, and Transport of Radioactivity

| Investigator | Person in Charge | Programs | Method of Reporting |
|---|---|---|---|
| Brookhaven National Laboratory | A. W. Castleman, Jr. | Release of fission products from fuel elements | Bimonthly and topical reports |
| Hanford Laboratories | E. R. Irish | Containment Systems Experiment (CSE) | Quarterly and topical reports |
| Harvard Air Cleaning Laboratory | L. Silverman | Fission-product removal | Topical reports |
| Oak Ridge National Laboratory | G. M. Watson and W. E. Browning, Jr. | Characterization and control of accident-released fission products | Semiannual progress reports and open literature |
| | G. M. Watson and C. J. Barton | Fission-product transport and retention | Semiannual progress reports |
| | G. M. Watson and W. E. Browning, Jr. | Innocuous simulation of accidents and hazards | Semiannual progress reports and topical reports |
| | G. M. Watson and G. W. Parker | Melting experiments in TREAT facility | Semiannual progress reports |
| | G. M. Watson and W. E. Browning, Jr. | Release of fission products upon in-pile melting of reactor fuels | Semiannual progress reports and open literature |
| | G. M. Watson and G. W. Parker | Release of fission products upon out-of-pile melting of reactor fuels | Semiannual progress reports and open literature |
| | W. B. Cottrell and L. F. Parsley | Nuclear Safety Pilot Plant (NSPP) | Semiannual progress reports and topical reports |
| | W. B. Cottrell and J. R. Buchanan | Nuclear Safety Information Center (NSIC) | State-of-the-art reports and staff studies |

### 12.5.1 Atomics International

The fission-product-retention properties of sodium are of interest in reactor systems using sodium as the coolant. An experimental program is being carried out at Atomics International in which emphasis is placed on investigating the inherent retention of iodine by sodium.

Reference

1. D. M. Elliott and W. P. Kunkel, Out-of-Pile Temperature Evaluation of Fission Product Retention in Sodium (900°–1400°F), USAEC Report NAA-SR-Memo-9383, Atomics International (undated).

### 12.5.2 Brookhaven National Laboratory

A program is in progress at Brookhaven for developing a fundamental understanding of the fission-product release mechanism and studying the manner by which release is interrelated with factors governing the chemical and physical state of the emanating fission products, as well as their transport behavior after release. The chemical and physical behavior of fission products released from uranium metal and uranium-molybdenum fuels into inert and oxidizing gaseous environments and from uranium metal and uranium oxide into steam was studied. Fission-product diffusion within the fuel, volatilization of fission products from the fuel, chemically enhanced release as a result of reactions between the fuel, the fission products, and the environment, together with deposition after release, are being studied with carbide fuels.

References

1. A. W. Castleman, F. E. Hoffmann, and A. M. Eshaya, Diffusion of Xenon Through Aluminum and Stainless Steel, USAEC Report BNL-624, Brookhaven National Laboratory, July 1960.
2. A. W. Castleman, F. E. Hoffmann, and A. M. Eshaya, Diffusion of Iodine Through Aluminum, USAEC Report BNL-644, Brookhaven National Laboratory, September 1960.
3. W. F. Kenney and A. M. Eshaya, Adsorption of Xenon on Activated Charcoal, USAEC Report BNL-689, Brookhaven National Laboratory, September 1960.
4. A. W. Castleman, Jr., The Chemical and Physical Behavior of Released Fission Products, Third Conference on Nuclear Reactor Chemistry, Oct. 9–11, 1962, pp. 155–168, USAEC Report TID-7641.
5. A. W. Castleman, Jr., and F. J. Salzano, Current Studies of Fission Product Behavior at BNL, Eighth AEC Air Cleaning Conference, October 1963, pp. 16–34, USAEC Report TID-7677.

6. A. W. Castleman, Jr., Chemical Considerations in Reactor Safety: The Chemical State of Fission Products Released from Uranium at High Temperature, Trans. Am. Nucl. Soc., 6(1): 128 (June 1963).

### 12.5.3 Hanford Laboratories

The Containment Systems Experiment (CSE) facility to be built at Hanford will, in addition to the capabilities mentioned in Section 12.3.3, be able to release simulated fission products within a model primary system. The amounts, rates, and times of release can be controlled during simulated loss-of-coolant accidents. The facility will be used to study the transport of the fission products within the primary system and the outer container and their leakage from the container. These phenomena will be studied as a function of internal pressure, concentration, type of aerosol, air atmosphere, air-steam atmosphere, and size of leak.

Iodine absorptive capacities of various metal and metal oxide surfaces and aerosolized solids are also being investigated.

References

None as yet.

### 12.5.4 Harvard Air Cleaning Laboratory

Studies of interest at the Harvard Air Cleaning Laboratory relate to the control of air contamination. The effectiveness and reliability of various filter media, particularly for iodine removal, are being determined. Investigations of a board of cell-like structure that could resist steam, pressure, and shock and serve as a porous filtration and adsorption membrane for released particulates and halogens are under way. The studies have also involved foams that can encapsulate particulates and halogens. If used inside a container during a reactor accident, the foam could presumably prevent much of the anticipated fission-product leakage from occurring.

References

1. L. Silverman, M. Corn, and F. Stein, Diffusion Board Containment Concepts and Foam Encapsulation Studies — A Progress Report, a paper delivered at Seventh AEC Air Cleaning Conference held at Brookhaven National Laboratory October 10–12, 1961, pp. 389–405, USAEC Report TID-7627, March 1962.

2. L. Silverman, Foam and Diffusion Board Approaches to Containment of Reactor Releases, a paper presented at the Third Conference on Nuclear Reactor Chemistry October 9–11, 1962, pp. 169–185, USAEC Report TID-7641, 1962.

3. R. Dennis and L. Silverman, Air Cleaning Studies - Progress Report, July 1, 1962 - December 31, 1962, USAEC Report HACL-96, Harvard University, June 1963.

4. R. E. Yoder, M. H. Fontana, and L. Silverman, Foam Suppression of Radioactive Iodine and Particulates, USAEC Report NYO-9324, Harvard University, March 1964.

### 12.5.5 Oak Ridge National Laboratory

Experiments are under way to investigate the fission-product release process in a systematic way. Involved are determinations of (1) the extent of fission-product release due to fuel oxidation, diffusion, or melting, (2) the extent of released fission-product deposition on reactor structural materials, (3) the fraction of fission products that can be removed from the atmosphere by use of various condensation, filtration, or other engineered techniques, and (4) the fraction and form of the fission products that escape from the containment system.

Experiments with both high- and low-burnup fuels are being conducted to determine whether burnup has an effect on the physical and chemical properties of the released fission products. The amount of the fission products released is varied to determine the effects of concentration on the behavior after release, such as aerosol agglomeration, chemical changes, and deposition. Both in-pile and out-of-pile experiments are being conducted to demonstrate the effect of method of melting on fission product behavior. The effects of high-temperature transport of fissioning fuel and of high-temperature surfaces surrounding the fuel are being studied, as well as the effects of environments of steam, noble gases, etc. Slow in-pile melting is accomplished at the Oak Ridge Research Reactor (ORR), and fast-transient melting is effected at the Transient Test Reactor Facility (TREAT) in Idaho in order to cover the range of melting rates that could occur in accidents.

The Nuclear Safety Pilot Plant (NSPP) at ORNL provides for nonnuclear heating, oxidation, meltdown, and vaporization of irradiated fuel elements and permits the study of fission-product release and transport on a sufficiently large geometrical scale to verify laboratory experiments. The transport of a broad enough range of fission products to bracket those that might be expected in a real accident will be studied in a model containment vessel under realistic conditions. The variables will include the type of carrier gas and convection current, type of atmosphere, and the amount, temperature, and chemical composition of available surfaces. Methods will be tested for the control of released fission products within containment vessels. These will include filters, water sprays, and submicron-particle conditioning by vapor condensation techniques.

The Nuclear Safety Information Center at Oak Ridge collects, evaluates, stores, and disseminates information on the containment of nuclear facilities and fission-product release, transport, and removal. Among the Center's functions is the preparation of staff studies of state-of-the-art reports on topics within the scope of its interest. Two of these reports are listed in the following references.

References

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3. C. E. Miller, Jr., R. P. Shields, B. F. Roberts, and W. E. Browning, Jr., Release of Fission Products by the In-Pile Melting of Uranium Dioxide Fuel, Trans. Am. Nucl. Soc., 5(1): 233 (June 1962).

4. G. W. Parker, G. E. Creek, and W. J. Martin, Release of Fission Products from Reactor-Grade $UO_2$ by Diffusion, Oxidation, and Melting, Proceedings of Nuclear Reactor Chemistry Conference, Gatlinburg, Tenn., October 10-12, 1961, pp. 149-159, USAEC Report TID-7622, July 1962.

5. Oak Ridge National Laboratory, Nuclear Safety Program Semiannual Progress Report for Period Ending June 30, 1962, USAEC Report ORNL-3319, Aug. 17, 1962.

6. W. E. Browning, Jr., and R. D. Ackley, Characterization of Millimicron Radioactive Aerosols and Their Removal from Gases, Third Conference on Nuclear Reactor Chemistry, Gatlinburg, Tenn., October 9-11, 1962, pp. 130-147, USAEC Report TID-7641.

7. B. F. Roberts, C. E. Miller, Jr., R. P. Shields, and W. E. Browning, Jr., Release of Fission Products by the In-Pile Destruction of Fuel, Third Conference on Nuclear Reactor Chemistry, Gatlinburg, Tenn., October 9-11, 1962, pp. 43-52, USAEC Report TID-7641.

8. G. W. Parker, G. E. Creek, R. H. Lorenz, and W. J. Martin, Parametric Studies of Fission-Product Release from $UO_2$ Fuels, Nuclear Reactor Chemistry Conference, Gatlinburg, Tenn., October 9-11, 1962, pp. 15-42, USAEC Report TID-7641.

9. Oak Ridge National Laboratory, Nuclear Safety Program Semiannual Progress Report for Period Ending December 31, 1962, USAEC Report ORNL-3401, Feb. 25, 1963.

10. R. E. Adams, C. J. Barton, W. E. Browning, Jr., and G. W. Parker, Particle and Fission-Product Behavior in Nuclear Accidents, Nucl. Safety, 4(3): 62-66 (March 1963).

11. Oak Ridge National Laboratory, Nuclear Safety Program Semiannual Progress Report for Period Ending June 30, 1963, USAEC Report ORNL-3483, Sept. 3, 1963.

12. M. H. Fontana and W. E. Browning, Effect of Particle Agglomeration on the Penetration of Filters Utilized with Double Containment Systems, USAEC Report ORNL-NSIC-1, Oak Ridge National Laboratory Nuclear Safety Information Center Report, Sept. 25, 1963.

13. L. F. Parsly, Jr., Nuclear Safety Pilot Plant Hazard Summary Report, USAEC Report ORNL-TM-683, Oak Ridge National Laboratory, Oct. 4, 1963.

14. P. P. Holz and C. M. Smith, Jr., Plasma Torch Design and Development for Nuclear Safety Pilot Plant Operations, USAEC Report ORNL-TM-687, Oak Ridge National Laboratory, Dec. 20, 1963.

15. K. E. Cowser, Current Practices in the Release and Monitoring of $I^{131}$ at NRTS, Hanford, Savannah River, and ORNL, USAEC Report ORNL-NSIC-3, Oak Ridge National Laboratory, Nuclear Safety Information Center Report, August 11, 1964.

16. Oak Ridge National Laboratory, Nuclear Safety Program Semiannual Progress Report for Period Ending December 31, 1963, USAEC Report ORNL-3547.

## 12.6 AIMS OF FUTURE CONTAINMENT RESEARCH PROGRAMS

Containment research programs have as a broad objective the determination of information on abnormal reactor behavior that will allow an analytical prediction of the effects of an accident on the reactor containment systems. These programs will not only establish the degree of reliance that can be placed on particular engineered safeguards but will also indicate the number and degree of these engineered features that are required for the location of a particular reactor and containment vessel on a particular site. It is therefore of interest to view the aims of the future AEC safety program as expressed by the chief of the AEC's Nuclear Safety Research and Development Branch at a recent meeting of the American Nuclear Society.* Material from this presentation forms the basis for the balance of this chapter.

In studying accident effects in a scientific manner, it is usual to start with small experimental rigs where a wide number of test variables can be conveniently evaluated. However, extrapolating from laboratory-scale data to full-scale reactor conditions cannot always be achieved with a high degree of confidence. It seems necessary to conduct experiments where scaling parameters can be evaluated by taking into account differences that exist between small and large masses, different surface-to-mass or surface-to-volume ratios, or different boundary conditions. Fuel subassembly mockups and even full-scale core tests of reactor accident effects appear necessary to confirm the theory and laws derived from the laboratory data. The final conviction that nuclear accident phenomena are predictable in their magnitude and controllable in terms of their ultimate consequences can best be demonstrated by the performance of experiments at suitable scaling levels.

Safety projects planned for the next few years will therefore include, in addition to laboratory-scale investigations, mockup and full-scale reactor experiments for the study of accident phenomena. A description of some of these plans follows.

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*S. A. Szawlewicz, Current Trends in Nuclear Safety Research and Development, a paper given at the annual meeting of American Nuclear Society, June 1963.

### 12.6.1 Transient Tests

Work on the study of energy releases and pressure generation from the meltdown of thin and thick aluminum-clad fuel plates will be performed using capsules driven by neutron bursts from a stainless-steel-clad-$UO_2$ core in SPERT I. The purpose of these tests is to get a better understanding of the SPERT I, SL-1, and BORAX I destructive pressure pulses. During the capsule experiments, pins like those of the oxide core will be encapsulated and taken to destruction to observe the maximum burst energy and minimum reactor period that are attainable without serious consequences and to determine what the threshold energy for violent pressure surges and fuel damage might be.

Other experiments directed to the fundamental understanding of the violent pressures associated with a nuclear excursion are conducted by the Space Technology Laboratories using the KEWB reactor to thermally drive a small uranium plate in a water-filled capsule in order to determine the dynamics of steam-void growth in reactor transients down to a 1-msec excursion period. The end objective is to be able to predict pressure transients in a violent excursion through the specification of surface-temperature history and knowledge of the rate at which new surface area is created when fuel is rapidly heated beyond the melting temperature.

### 12.6.2 Metal-Water Reactions

Most of the fundamental data required for the prediction of the rate and extent of metal-water reactions during postulated reactor accidents have been obtained, although some work on the effect of pressure on reaction rate remains to be done. However, there appears to be no practical theoretical approach to the problem of predicting what particle or droplet sizes will be produced during core destruction or their rate of formation. Thus an extensive experimental program of particle-size measurements is under way that involves in-pile meltdown of specimens under various accident conditions. Most of this work is now carried out in the TREAT reactor. Since the TREAT tests are limited to relatively small sample sizes and to reactor periods of 40 msec or longer, additional experiments are planned to extend the data to periods of 3 to 4 msec using larger scale pin-cluster specimens. These tests will be carried out in the Power Burst Facility (see below). It is anticipated that large-scale simulations of loss-of-coolant accidents (with slow heatup) will also be carried out in the Power Burst Facility.

### 12.6.3 Power Burst Facility

Extension of accident-effects studies to large fuel subassemblies subjected to short damaging nuclear excursions will be possible with the development of an advanced pulsed-neutron reactor named the Power Burst Facility (PBF). The objective is to obtain a core with a self-limiting excursion response to reactor periods as short as 1 msec. A cylindrical region 8 in. in diameter in the center of the core would be fitted with an experimental loop to accommodate fuel test samples consisting of rod bundles, plates, or other power-reactor configurations, including thermal- and fast-reactor mockups. The Power Burst Facility would permit experimental study of the microscopic and gross effects associated with slow or rapid heating to meltdown of reactor fuels subjected to a power transient. Scaling effects could be studied directly by melting small samples, initially, and extending the tests to include large fuel clusters having mass-to-coolant ratios and geometric configurations equivalent to those of a full-scale reactor.

### 12.6.4 Fission-Product Release and Large-Scale Tests

Considerable information has been obtained on the fractional release of fission isotopes from small fuel samples with many different test variables, such as temperature of melt, time at melt temperature, burnup, type of environment, type of fuel, and effect of cladding.

The work to date has largely been performed with samples in the 1-mg to 39-g range and burnups in some cases up to 11,000 Mwd/ton. Melting has for most tests been accomplished by electrical induction heating or by an arc-image furnace, but recently work has been started with the use of electrical resistance and in-core heating of capsules in the Oak Ridge Research Reactor at ORNL (see sec. 12.5.4). At Brookhaven National Laboratory, other experiments are being performed to obtain an understanding of reaction kinetics at the time of fuel melting and to provide a basis for predicting the chemical and physical form of the released fission products in reactor accidents (see sec. 12.5.1). Work at Atomics International concerns the behavior of fission products released in a sodium environment (see sec. 12.4.2).

Extension of testing to other than small samples and to conditions more closely approximating those of real meltdowns is now planned. Meltdown experiments for studying fission-product-release behavior have recently been started at ORNL with a cluster of seven small fuel pins. Reductions in the release of iodine by a factor of 2 compared with the release in the small sample meltdowns have been noted. Capsule tests have been performed in TREAT to determine the release of activity due to transient nuclear heating. These tests have yielded significant results in that they have shown that the amounts of activity released are much less than in out-of-pile or other in-pile tests where the temperature transient is much slower and longer. When the Power Burst Facility is completed, fission-product releases from the meltdown of large fuel clusters will be studied.

For pressurized reactors, the maximum credible accident is currently assumed to start with the complete severance or rupture of the primary coolant line, leading to loss of coolant, possible core meltdown, and release of activity to the containment. Both the portion of released activity that remains available for leakage to the atmosphere and the internal postaccident pressures available for driving such leakage are of considerable interest. In the Nuclear Safety Pilot Plant at ORNL, relatively large fuel specimens of all types up to 1000 curies of activity will be melted to determine the natural transport effects, such as plateout, agglomeration, and diffusion, in a 1500-$ft^3$ simulated containment vessel. In the vessel it will also be possible to examine some of the engineering safeguards referred to earlier, such as the effectiveness of sprays, steam, or other scavengers in removing airborne activity from the containment atmosphere as a function of time, concentration, aerosol size, etc. Air recirculation systems using charcoal-filter combinations will also be studied.

To study the consequences of reactor accidents further, particularly for the water-cooled systems of major current interest, a multipurpose containment test facility called the Containment System Experiment (CSE) is being built (see secs. 12.3.3 and 12.5.2). The facility is essentially a nonnuclear simulator of pressurized- and boiling-water systems, in which the dynamics of vessel blowdown, the pressure-time history with and without suppression systems, leakage measurements, and transport of simulated fission products can be determined. Comparison of real and simulated fission-product releases will first be made in NSPP to develop fission-product simulants that can safely be handled in the larger CSE.

The CSE, in addition to its ability to perform "short-term" tests for quick demonstration of containment capabilities, can be used to perform "long-term" parametric studies leading to optimization of containment characteristics. The facility can simulate the coolant inventories and initial pressures and temperature of boiling- and pressurized-water reactors up to 1000 Mw electrical output at about one-fifth linear scale and be used to mock up the maximum credible accident for water reactors, i.e., the double-ended rupture of a main coolant pipe which would lead to sudden loss of coolant. In addition, a range of ruptures of the primary system smaller than the maximum credible rupture and up to twice the maximum credible rupture can be developed to examine the trend of containment response and the margins of safety inherent in design. The release of simulated fission products within the model's primary system would also be undertaken in amounts, at rates, and at times during the loss of coolant sequence that could bracket ranges of these values in operating reactors. The transport of such activity within the primary system and within the outer containment shell, with leakage as a function of internal pressure, concentration and type of aerosol, type of atmosphere, and size of leak could also be studied over wide ranges of conditions. Leakage could be simultaneously measured by several different techniques whose relative accuracy and desirability could then be determined.

Another containment study that was started recently deals with the problem of pipe rupture in pressurized reactor-coolant circuits. Pipe rupture is usually postulated as the initiating event that leads to core uncovering, core meltdown, and fission-product release for water reactors. An estimate of the probability, the maximum size of rupture, the crack growth rate, and the initiating mechanisms for pipe failure can be determined and understood by a suitable research program; this can lead to design measures and materials inspection that could readily be applied to the prevention of such accidents. The mechanisms of brittle and ductile pipe failure for pressurized-reactor plants will be studied with experimental loops by subjecting pipe sections to cyclic loads of temperature, pressure, bending moments, and other stresses simulating unusual operating conditions.

Care will be taken to characterize the metallurgy and physical strength of the piping material properly before starting the test cycles. The effect of materials defects upon crack initiation and propagation will be observed. Critical zones to be examined will include nozzles, bends, transition zones, welds, etc.

Full-scale tests of reactor-accident effects are being planned by the Nuclear Safety Engineering Test Branch in the AEC's Division of Reactor Development. They will be conducted in the LOFT facility (loss-of-flow tests) which will be operated for the AEC by the Phillips Petroleum Company at the National Reactor Test Site in Idaho (see sec. 12.3.4).

LOFT will be used to conduct loss-of-coolant tests leading to the ultimate meltdown of the core for a 50-Mw(thermal) pressurized-water reactor. A full-scale containment shell constitutes a part of this facility.

### 12.6.5 Conclusion

Nuclear-safety projects sponsored by the Commission during the coming years will emphasize reactor-accident modeling, prevention, and control. The programs planned are based on the awareness that the causes, consequences, and fears associated with major nuclear accidents can best be resolved by undertaking large-scale and engineering-type experiments to complement the basic research programs undertaken in the laboratory.

The transition from laboratory level to large-scale experimental tests of accident effects is a necessary step that must be taken to place safety in design, siting, and hazards evaluation upon a firmer technical footing.

## Appendix A

## CODE OF FEDERAL REGULATIONS, TITLE 10
## PART 100 - REACTOR SITE CRITERIA

Pursuant to the Administrative Procedures Act and the Atomic Energy Act of 1954, as amended, the following guide is published as a document subject to codification, to be effective 30 days after publication in the FEDERAL REGISTER.

Statement of considerations. On February 11, 1961, the Atomic Energy Commission published in the FEDERAL REGISTER a notice of proposed rule making that set forth general criteria in the form of guides and factors to be considered in the evaluation of proposed sites for power and testing reactors. The Commission has received many comments from individuals and organizations, including several from foreign countries, reflecting the widespread sensitivity and importance of the subject of site selection for reactors. Formal communications have been received on the published guides, including a proposed comprehensive revision of the guides into an alternate form.

In these communications, there was almost unanimous support of the Commission's proposal to issue guidance in some form on site selections, and acceptance of the basic factors included in the proposed guides, particularly in the proposal to issue exposure dose values which could be used for reference in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence.

On the other hand, many features of the proposed guides were singled out for criticism by a large proportion of the correspondents. This was particularly the case for the appendix section of the proposed guides, in which was included an example calculation of environmental distance characteristics for a hypothetical reactor. In this appendix, specific numerical values were employed in the calculations. The choice of these numerical values, in some cases involving simplifying assumptions of highly complex phenomena, represent types of considerations presently applied in site calculations and result in environmental distance parameters in general accord with present siting practice. Nevertheless, these particular numerical values and the use of a single example calculation were widely objected to, basically on the grounds that they presented an aspect of inflexibility to the guides which otherwise appeared to possess considerable flexibility and tended to emphasize unduly the concept of environmental isolation for reactors with minimum possibility being extended for eventual substitution thereof of engineered safeguard.

In consequence of these many comments, criticisms and recommendations, the proposed guides have been rewritten, with incorporation of a number of suggestions for clarification and simplification, and elimination of the numerical values and example calculation formerly constituting the appendix to the guides. In lieu of the appendix, some guidance has been incorporated in the text itself to indicate the considerations that led to establishing the exposure values set forth. However, in recognition of the advantage of example calculations in providing preliminary guidance to application of the principles set forth, the AEC will publish separately in the form of a technical information document a discussion of these calculations.

These guides and the technical information document are intended to reflect past practice and current policy of the Commission of keeping stationary power and test reactors away from densely populated centers. It should be equally understood, however, that applicants are free and indeed encouraged to demonstrate to the Commission the applicability and significance of considerations other than those set forth in the guides.

One basic objective of the criteria is to assure that the cumulative exposure dose to large numbers of people as a consequence of any nuclear accident should be low in comparison with what might be considered reasonable for total population dose. Further, since accidents of greater potential hazard than those commonly postulated as representing an upper limit are conceivable, although highly improbable, it was considered desirable to provide for protection against excessive exposure doses to people in large centers, where effective protective measures might not be feasible. Neither of these objectives were readily achievable by a single criterion. Hence the population center distance was added as a site requirement when it was found for several projects evaluated that the specification of such a distance requirement would approximately fulfill the desired objectives and reflect a more accurate guide to current siting practices. In an effort to develop more specific guidance on the total man-dose concept, the Commission intends to give further study to the subject. Meanwhile, in some cases where very large cities are involved, the population center distance may have to be greater than those suggested by these guides.

A number of comments received pointed out that AEC siting factors included considerations of population distributions and land use surrounding proposed sites but did not indicate how future population growth might affect sites initially approved. To the extent possible, AEC review of the land use surrounding a proposed site includes considerations of potential residential growth. The guides tend toward requiring sufficient isolation to preclude any immediate problem. In the meantime, operating experience that will be acquired from plants already licensed to operate should provide a more definitive basis for weighing the effectiveness of engineered safeguards versus plant isolation as a public safeguard.

These criteria are based upon a weighing of factors characteristic of conditions in the United States and may not represent the most appropriate procedure nor optimum emphasis on the various interdependent factors involved in selection of sites for reactors in other countries where national needs, resources, policies and other factors may be greatly different.

Sec.

SITE EVALUATION FACTORS



AUTHORITY: Pars. 100.1 to 100.11 issued under sec. 103, 68 Stat. 936, sec. 104, 68 Stat. 937, sec. 161, 68 Stat. 948, sec. 182, 68 Stat. 953; 42 U.S.C. 2133, 2134, 2201, 2232.

SOURCE: Pars. 100.1 to 100.11 appear at 27 F.R. 3509, Apr. 12, 1962.

100.1 Purpose. (a) It is the purpose of this part to describe criteria which guide the Commission in its evaluation of the suitability of proposed sites for stationary power and testing reactors subject to Part 50 of this chapter.

(b) Insufficient experience has been accumulated to permit the writing of detailed standards that would provide a quantitative correlation of all factors significant to the question of acceptability of reactor sites. This part is intended as an interim guide to identify a number of factors considered by the Commission in the evaluation of reactor sites and the general criteria used at this time as guides in approving or disapproving proposed sites. Any applicant who believes that factors other than those set forth in the guide should be considered by the Commission will be expected to demonstrate the applicability and significance of such factors.

100.2 Scope. (a) This part applies to applications filed under Part 50 and 115 of this chapter for stationary power and testing reactors.

(b) The site criteria contained in this part apply primarily to reactors of a general type and design on which experience has been developed, but can also be applied to other reactor types. In particular, for reactors that are novel in design and unproven as prototypes or pilot plants, it is expected that these basic criteria will be applied in a manner that takes into account the lack of experience. In the application of these criteria which are deliberately flexible, the safeguards provided - either site isolation or engineered features - should reflect the lack of certainty that only experience can provide.

100.3 Definitions. As used in this part:

(a) "Exclusion area" means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result.

(b) "Low population zone" means the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident. These guides do not specify a permissible population density or total population within this zone because the situation may vary from case to case. Whether a specific number of people can, for example, be evacuated from a specific area, or instructed to take shelter, on a timely basis will depend on many factors such as location, number and size of highways, scope and extent of advance planning, and actual distribution of residents within the area.

(c) "Population center distance" means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

(d) "Power reactor" means a nuclear reactor of a type described in Par. 50.21(b) or 50.22 of this chapter designed to produce electrical or heat energy.

(e) "Testing reactor" means a "testing facility" as defined in Par. 50.2 of this chapter.

## SITE EVALUATION FACTORS

100.10 Factors to be considered when evaluating sites. Factors considered in the evaluation of sites include those relating both to the proposed reactor design and the characteristics peculiar to the site. It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in release of significant quantities of radioactive fission products. In addition, the site location and the engineered features included as safeguards against the hazardous consequences of an accident, should one occur, should insure a low risk of public exposure. In particular, the Commission will take the following factors into consideration in determining the acceptability of a site for a power or testing reactor:

(a) Characteristics of reactor design and proposed operation including:

(1) Intended use of the reactor including the proposed maximum power level and the nature and inventory of contained radioactive materials;

(2) The extent to which generally accepted engineering standards are applied to the design of the reactor;

(3) The extent to which the reactor incorporates unique or unusual features having a significant bearing on the probability or consequences of accidental release of radioactive materials;

(4) The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur.

(b) Population density and use characteristics of the site environs, including the exclusion area, low population zone, and population center distance.

(c) Physical characteristics of the site, including seismology, meteorology, geology and hydrology.

(1) The design for the facility should conform to accepted building codes or standards for areas having equivalent earthquake histories. No facility should be located closer than one-fourth mile from the surface location of a known active earthquake fault.

(2) Meteorological conditions at the site and in the surrounding area should be considered.

(3) Geological and hydrological characteristics of the proposed site may have a bearing on the consequences of an escape of radioactive material from the facility. Special precautions should be planned if a reactor is to be located at a site where a significant quantity of radioactive effluent might accidentally flow into nearby streams or rivers or might find ready access to underground water tables.

(d) Where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards.

100.11 Determination of exclusion area, low population zone, and population center distance. (a) As an aid in evaluating a proposed site, an applicant should assume a fission product release[1] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:

(1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem[2] or a total radiation dose in excess of 300 rem[2] to the thyroid from iodine exposure.

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[1]The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

[2]The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP recommendations may be disregarded in the determination of their radiation exposure status (see NBS Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

(2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least one and one-third times the distance from the reactor to the outer boundary of the low population zone. In applying this guide, due consideration should be given to the population distribution within the population center.

Where very large cities are involved, a greater distance may be necessary because of total integrated population dose consideration.

(b) For sites for multiple reactor facilities consideration should be given to the following:

(1) If the reactors are independent to the extent that an accident in one reactor would not initiate an accident in another, the size of the exclusion area, low population zone and population center distance shall be fulfilled with respect to each reactor individually. The envelopes of the plan overlay of the areas so calculated shall then be taken as their respective boundaries.

(2) If the reactors are interconnected to the extent that an accident in one reactor could affect the safety of operation of any other, the size of the exclusion area, low population zone and population center distance shall be based upon the assumption that all interconnected reactors emit their postulated fission product releases simultaneously. This requirement may be reduced in relation to the degree of coupling between reactors, the probability of concomitant accidents and the probability that an individual would not be exposed to the radiation effects from simultaneous releases. The applicant would be expected to justify to the satisfaction of the AEC the basis for such a reduction in the source term.

(3) The applicant is expected to show that the simultaneous operation of multiple reactors at a site will not result in total radioactive effluent releases beyond the allowable limits of applicable regulations.

NOTE: For further guidance in developing the exclusion area, the low population zone, and the population center distance, reference is made to Technical Information Document 14844, dated March 23, 1962, which contains a procedural method and a sample calculation that result in distances roughly reflecting current siting practices of the Commission. The calculations described in Technical Information Document 14844 may be used as a point of departure for consideration of particular site requirements which may result from evaluation of the characteristics of a particular reactor, its purpose and method of operation.

Copies of Technical Information Document 14844 may be obtained from the Commission's Public Document Room, 1717 H Street NW., Washington, D.C., or by writing the Director, Division of Licensing and Regulation, U.S. Atomic Energy Commission, Washington 25, D.C.

# Appendix B

## DESCRIPTION OF CONTAINMENT SYSTEM IN FINAL SAFETY ANALYSIS REPORT

The containment system description in the final safety analysis report should provide information appropriate to a determination of the effectiveness of the containment system in limiting the release of radioactive materials. To the extent that they may be applicable, the following guides are suggested as illustrative of the types of information that should be considered for inclusion in this section:*

A. Description of Containment System. A general description of the containment system, including, for example:

(1) A description of the design bases for the containment system including a concise discussion of any provisions for the venting, suppression, or reduction of pressure.
(2) A description of the circumstances and conditions under which the containment system is to be sealed from the atmosphere, the methods used to detect these circumstances and conditions, and the sequence of events following the signal for closure, including the operation of closure devices and actuation of emergency systems.
(3) A statement as to the principal dimensions and approximate gross and net volumes of the containment system.
(4) A discussion, preferably supported by sketches, describing the containment system, and service and auxiliary facilities.
(5) The extent of occupancy of the containment system which will be permitted during operation, maintenance, testing and experimentation; and measures to control such occupancy.
(6) An explanation of any activities, other than reactor operation, that are to be conducted within the containment system.
(7) The extent to which the containment system integrity is compromised, if at all, during refueling or other maintenance activities, and the condition of the reactor, including degree of shutdown, and temperature and pressure levels during these periods.
(8) A discussion of any planned limitations on operation or maintenance that are prohibited whenever the design integrity of the containment system is not established.

B. Design Loading. A tabulation of the design loading for the containment vessel including, for example:

(1) The internal pressure load.

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*As taken from Aug. 28, 1962, draft of AEC Licensing Guide entitled "Purpose, Organization and Contents of Hazards Summary Reports for Power Reactors."

(2) Thermal loads, including transient loads occurring in the event of accident, and due to condensation of hot vapors on internal surfaces, heat transfer through the system walls, and temperature differentials that may exist at points of embedment or restraint.
(3) Concentrated loads, such as:
  (a) Impact loads from internal credible missiles;
  (b) Loads due to supporting members, components, and equipment connected to the system;
  (c) Live loads from cranes and floor loadings transmitted directly to the containment system walls;
  (d) The dead load due to transfer of the containment system weight to the supporting structure;
  (e) Loadings from internal flooding of the system.
(4) External loads, such as:
  (a) Seismic effects;
  (b) Loads due to nonuniform settling of foundations;
  (c) Wind, snow, and ice loads;
  (d) Vacuum conditions due to effects of barometric and ambient temperature fluctuations on a sealed system;
  (e) Vacuum conditions due to internal depletion of oxygen through combustion;
  (f) Loadings from external soil and hydrostatic pressures;
  (g) Impact loads from external credible missiles.

C. <u>Materials of Construction</u>. A description of the materials of construction, including, for example:

(1) A list of applicable materials specifications for the principal materials of construction.
(2) The extent to which amendments, supplements, or waivers have been applied to standard material specifications, including any modification of the provisions for heat treatment, chemical composition, mechanical properties, or material inspections, tests, and inspection standards.
(3) A description of notch-ductility properties, including:
  (a) Results of Charpy V-notch impact tests on vessel material, weld specimens, and component weldments; specified at particular test temperatures;
  (b) A description of any heat treatment applied to enhance the notch-ductility properties.
(4) A description of corrosion protection features, including material allowances, provided for the containment system.

D. <u>Structural Design</u>. A description of structural design criteria and features, including, for example:

(1) Codes observed in the design and construction of the containment system, and any exceptions, revisions, addenda, case interpretations, or special rulings by which the application of codes has been qualified.
(2) The design stress limits for combinations of primary and secondary stress in terms of the allowable stress value for the containment system vessel material.

(3) The method of supporting the containment system, including the means employed in transferring the vertical dead loads from within the system to the foundation.
(4) Containment system drawings in sufficient detail to include:
  (a) Location and identification of major components within the system;
  (b) Basic system dimensions and wall thicknesses;
  (c) Allowable deviation from circular or spherical form;
  (d) Geometrical configuration of the system;
  (e) Identification, location, and size of connections and penetrations;
  (f) Internal or external attachments;
  (g) Vessel supporting structure;
  (h) Location of major welding joints;
  (i) Transitions between sections of unequal thickness;
  (j) Grade level;
  (k) Areas of embedment in concrete, and relation to grade level.
(5) A brief description of fabrication and welding procedures, and of specifications for shop and site fabrication and of welding employed in the containment system manufacture and not included in applicable codes or specifications.
(6) Inspection procedures, such as:
  (a) Material inspections;
  (b) Inspections during shop fabrication;
  (c) Inspections during site erection;
  (d) Weld operator qualifications inspections;
  (e) Welding procedures inspections;
  (f) Heat treatment practices;
  (g) Application of, and acceptance standards and procedures for radiographic, ultrasonic, magnetic particle, and liquid penetrant inspections;
  (h) Records of inspections.
(7) A copy of the vessel data report prepared by the manufacturer of the containment system vessel.
(8) Details of the design and sealing of joints in the containment system.
(9) Provisions made to integrate piping and ducting penetrations with the containment system wall in order to preclude a rupture between the wall and the closure valves provided for the penetrating lines, or of the wall itself.
(10) The dimensions, location, methods of support, and materials of construction of any secondary or containment radiation shielding associated with the containment system, and of any thermal or weather insulation.
(11) The dimensions, location, methods of support, and materials of construction of any missile barriers associated with the containment system.
(12) Uses made, if any, of the containment system structure for support of equipment.

E. Penetrations. An analysis of penetrations of the containment system, including, for example:

(1) The numbers and types of penetration of the containment system for the entry or exit of electrical wiring, fluid piping, tubing, and ducts, and all airlocks and access ports.
(2) Significant details of design, construction, and operation of penetrations, including:
 (a) Location and grouping of penetrations;
 (b) Methods and materials for sealing penetrations to the containment system walls and anticipated requirements of maintenance, including removal and replacement of the sealing.
(3) A description of valves and dampers in pipes, tubes, or ducts penetrating the containment system, including:
 (a) The locations, and types of valves and dampers;
 (b) The closing times of valves and dampers;
 (c) Method of actuation of valves and dampers.
(4) A description of airlocks and access ports penetrating the containment system, including, for example:
 (a) Dimensions of airlocks and access ports;
 (b) Operating methods for airlocks and access ports;
 (c) Frequency of operation of airlocks and access ports;
 (d) Interlock and bypass provisions in connection with airlocks and access ports;
 (e) A detailed description of any emergency escape ports and the occasions for which they are provided.
(5) A description of penetrations for air sampling, and connections for maintenance services.
(6) A description of any temperature, pressure, humidity, radiation, or chemical concentration detection or monitoring system used in conjunction with penetrations.
(7) A description of the methods, minimum acceptance standards, and frequency of inspection and testing of components and systems essential to reliable operation of the containment system in event of an accident, including:
 (a) Automatically actuated valves and dampers;
 (b) Detection and monitoring system;
 (c) Airlocks;
 (d) Spray systems;
 (e) Emergency cooling systems.

F. Ventilation. A description of provisions for ventilation of the containment system, and for other air purification facilities servicing the containment system under normal and emergency conditions, including, for example:

(1) Method of ventilation, whether operation is continuous or intermittent, and the basis for selection of the type of operation;
(2) Ventilation flow rates and the basis for selection of these flow rates;
(3) The flow paths of ventilating air, major pressure gradients, and the location of fans and blowers;

(4) A description of filter systems and other air purification equipment, including types of equipment, purpose, location and efficiencies of filters as installed, testing of filter efficiency, inspection and servicing requirements, and accessibility.

(5) Requirements of, and provisions for, separate ventilation of the shielding, or any other individual component or system within the containment system, including the relation of this ventilating system to the main containment system ventilating system.

(6) Maximum anticipated rate of heat release to the containment system air, during normal operation, from systems and equipment installed within the containment system.

(7) A description of the air conditions relative to temperature, humidity, and radioactivity to be maintained in the containment system during periods of normal operation, and of the air treatment systems provided to maintain these conditions, and to dehumidify stagnant regions where corrosion is of concern.

(8) A description of the facilities provided and the methods used to exhaust, monitor, and filter the ventilation air from the containment system in a safe manner, including provisions made to disperse the exhaust so as to preclude re-entry to the facility through any air intake.

(9) A description of design features potentially capable of alleviating the hazards associated with an accident, including any provisions for recirculation of the containment system air through a filter unit following an accident releasing radioactivity to the containment system.

G. Design Pressure. An analysis of the containment system internal design pressure, including, for example:

(1) The basis of choice of the containment system internal design pressure value, including, for example:

(a) Sources and amounts of energy and material released to the containment system as the result of credible ruptures of various sized pipes in the primary coolant system, and the secondary coolant system if a rupture therein could credibly involve the primary system, the time dependencies associated with these releases, justification for the type of release selected for design purposes, and a comparison of the effects resulting from this release with those which would result from a release due to a circumferential rupture of the largest pipe in the primary coolant system with both ends of this pipe, at the point of rupture, free and open;

(b) Mechanisms of energy absorption and transfer, and the associated time dependency of the absorption and transfer;

(c) Graphical presentation of the pressure and temperature within the containment system as a function of time following the assumed maximum credible accident, with and without emergency cooling and spray systems functioning, and extended in time to include all maxima;

(d) Discussion of all principal pressure and temperature maxima, including their cause and effect;
(e) A description and analysis of any conditions that could subject the containment system to a negative pressure, the provisions made for relief of negative pressures, the means of protection against exterior blocking of the relief line, and the methods, and frequency of periodic testing of the relief mechanism.

(2) Pressure tests and inspections for structural integrity, including:
(a) Methods of pressure testing the containment system prior to routine reactor operation, and pressures at which testing will be conducted;
(b) Description of the methods for periodic retesting for strength after routine operation of the reactor;
(c) Methods and frequency of inspection of the containment system for structural integrity, including the extent and standards of inspection applied, and the extent to which both sides of the containment system walls are accessible for inspection.

H. Design Leakage Rate. An analysis of the containment system design leakage rate, including, for example:

(1) The basis of choice of the containment system design leakage rate, including:
(a) The design leakage rate;
(b) The relationship between leakage rate and containment system and environmental pressure.

(2) Leak rate tests, including:
(a) Description of leak rate tests to be performed prior to routine reactor operation stating for each test, the extent of construction completed, the condition of penetrations, the pressure at which the test will be conducted, the limitations of the allowable leak rate and a description of the method of testing, including the duration, precision, and accuracy of tests, and corrections made in the reduction of test data to final values;
(b) Description of the methods of retesting for leak tightness after routine operation of the reactor, including frequency, duration, precision, and accuracy of tests, the methods of data reduction, and provisions made for leak testing individual penetrations.

I. Miscellaneous. A discussion, including appropriate descriptions and analyses, of miscellaneous features and considerations associated with the containment system, including, for example:

(1) The methods and criteria used in the stress analysis of the containment system.
(2) An analysis of sources of missiles having a capability of damaging the containment system, the primary coolant and associated

high pressure auxiliary systems, or emergency systems essential to the integrity of the containment system.

(3) A description of the types, amounts, locations, and characteristics of materials, in and about the containment system, that are inflammable or explosive in nature under the conditions that may prevail during normal operation and in event of accident.

(4) A description of design features provided to prevent or capable of preventing penetration of the containment system by the worst credible core meltdown.

(5) A description of design features provided for decontamination of the containment system and of equipment therein under normal operating conditions and in event of accident.

(6) Provisions for preserving the integrity of the containment system in the event of fire, flood, electrical storm, earthquake, or other emergency, including, for example:

- (a) An analysis of the extent to which such provisions are required;
- (b) Systems and equipment to be employed;
- (c) Methods of actuation of these systems and equipment, and an analysis of the adequacy of the methods of actuation;
- (d) Programs for establishing the reliability of, and for testing these systems.

(7) A description of the means provided for determination of post-accident conditions within the containment system, including pressure, temperature, and radioactivity levels, and the status of critical components and systems.

(8) A description of provisions made to maintain equipment, and components within the containment system in a fail-safe condition following an accident, including protection of valve operators against collapse due to pressure, protection against wet cable runs, protection against wrong-way operation of air actuated valves due to pressure, and protection of essential emergency equipment against damage.

(9) Provisions for secondary methods of actuation, including manual actuation, of safety devices, components, equipment, and systems essential to the continued integrity of the containment system in the event of accident, and the times required for such emergency actuation under conditions when normal actuation does not occur.

(10) A description of the containment system communication facilities.

# Appendix C

# CRITERIA OF SECTION III OF THE ASME BOILER AND PRESSURE VESSEL CODE FOR NUCLEAR VESSELS

# CRITERIA OF SECTION III OF THE ASME BOILER AND PRESSURE VESSEL CODE FOR NUCLEAR VESSELS

## DESIGN

### I. INTRODUCTION

The design philosophy of the present Section I (Power Boilers) and Section VIII (Unfired Pressure Vessels) of the ASME Boiler Code may be inferred from a footnote which appears in Section VIII on page 8 of the 1962 edition. This footnote refers to a sentence in Par. UG-23 (c) which states, in effect, that the wall thickness of a vessel shall be such that the maximum hoop stress does not exceed the allowable stress. The footnote says:

> "It is recognized that high localized and secondary bending stresses may exist in vessels designed and fabricated in accordance with these rules. Insofar as practical, design rules for details have been written to hold such stresses at a safe level consistent with experience."

What this means is that Sections I and VIII do not call for a detailed stress analysis but merely set the wall thickness necessary to keep the basic hoop stress below the tabulated allowable stress. They do not require a detailed evaluation of the higher, more localized stresses which are known to exist, but instead allow for these by the safety factor and a set of design rules. Examples of such rules are the minimum allowable knuckle radius for a torispherical head and the "area replacement" rules for reinforcement of openings. Thermal stresses are given even less consideration. The only reference to them is in Par. UG-22 where "the effect of temperature gradients" is listed among the loadings to be considered. There is no indication of how this consideration is to be given. On the other hand, the Piping Code (ASA-B31.1) does give allowable values for the thermal stresses which are produced by the expansion of piping systems and even varies these allowable stresses with the number of cycles expected in the system.

The Special Committee to Review Code Stress Basis was originally established to investigate what changes in Code design philosophy might permit use of higher allowable stresses without reduction in safety. It soon became clear that one approach would be to make better use of modern methods of stress analysis. Detailed evaluation of actual stresses would permit substituting knowledge of localized stresses, and assignment of more rational margins, in place of a larger factor which really reflected lack of knowledge.

The simplified procedures of Section VIII may be in error either on the side of overconservatism or on the side of being inapplicable for the more severe types of service. Detailed analysis of almost any Code vessel would show where the design could be optimized to conserve material. On the other hand, vessels designed to minimum Section VIII standards may not be suitable for highly cyclic types of operation or for nuclear service where periodic inspection is usually difficult and sometimes impossible. It was these thoughts which led to the writing of the Nuclear Cases, particularly N-1272 and N-1273, and finally to the preparation of Section III.

The development of analytical and experimental techniques have made it possible to determine stresses in considerable detail. When the stress picture is brought into focus, it is not reasonable to retain the same values of allowable stress for the clear detailed picture as had previously been used for the less detailed one. Neither is it sufficient merely to raise the allowable stresses to reasonable values for the peak stresses, since peak stress by itself is not an adequate criterion of safety. A calculated value of stress means little until it is associated with its location and distribution in the structure and with the type of loading which produced it. Different types of stress have different degrees of significance and must, therefore, be assigned different allowable values. For example, the average hoop stress through the thickness of the wall of a vessel due to internal pressure must be held to a lower value than the stress at the root of a notch in the wall. Likewise, a thermal stress can often be allowed to reach a higher value than one which is produced by dead weight or pressure. Therefore the Special Committee developed a new set of design criteria which shifted the emphasis away from the use of standard configurations and toward the detailed analysis of stresses. The setting of allowable stress values required dividing stresses into categories and assigning different allowable values to different groups of categories. These criteria were used in preparing Section III, Vessels in Nuclear Service, and will be described here.

## Definitions

When discussing various combinations of stresses produced by various types of loading, it is important to use terms which are clearly defined. For example, the terms "membrane stress" and "secondary stress" are often used somewhat loosely. However, when a limit is to be placed on membrane stress, it is imperative that there must be no question about what is meant. Therefore the Special Committee spent a considerable amount of time in preparing a set of definitions. These definitions are given in Par. N-412 of Section III.

## Strength Theories

The stress state at any point in a structure may be completely defined by giving the magnitudes and directions of the three principal stresses. When two or three of these stresses are different from zero, the proximity to yielding must be determined by means of a strength theory. The theories most commonly used are the maximum stress theory, the maximum shear stress theory (also known as the Tresca criterion), and the distortion energy theory (also known as the octahedral shear theory and the Mises criterion). It has been known for many years that the maximum shear stress theory and the distortion energy theory are both much better than the maximum stress theory for predicting both yielding and fatigue failure in ductile metals. Sections I and VIII use the maximum stress theory, by implication, but Section III uses the maximum shear theory. Most experiments show that the distortion energy theory is even more accurate than the shear theory, but the shear theory was chosen because it is a little more conservative, it is easier to apply, and it offers some advantages in some applications of the fatigue analysis, as will be shown later.

The maximum shear stress at a point is defined as one-half of the algebraic difference between the largest and the smallest of the three principal stresses. Thus, if the principal stresses are $\sigma_1$, $\sigma_2$, and $\sigma_3$, and $\sigma_1 > \sigma_2 > \sigma_3$ (algebraically), the maximum shear stress is $\frac{1}{2}(\sigma_1 - \sigma_3)$. The maximum shear stress theory of failure states that yielding in a component occurs when the maximum shear stress reaches a value equal to the maximum shear stress at the yield point in a tensile test. In the tensile test, at yield, $\sigma_1 = S_y$, $\sigma_2 = 0$, and $\sigma_3 = 0$; therefore the maximum shear stress is $S_y/2$. Therefore yielding in the component occurs when

$$\tfrac{1}{2}(\sigma_1 - \sigma_3) = \tfrac{1}{2} S_y \,. \tag{1}$$

In order to avoid the unfamiliar and unnecessary operation of dividing both the calculated and the allowable stresses by two before comparing them, a new term called "equivalent intensity of combined stress" or, more briefly, "stress intensity" has been used. The stress intensity is defined as twice the maximum shear stress and is equal to the largest algebraic difference between any two of the three principal stresses. Thus the stress intensity is directly comparable to strength values found from tensile tests.

For the simple analyses on which the thickness formulas of Section I and VIII are based, it makes little difference whether the maximum stress theory or the maximum shear stress theory is used. For example, in the wall of a thin-walled cylindrical pressure vessel, remote from any discontinuities, the hoop stress is twice the axial stress and the radial stress on the inside is compressive and equal to the internal pressure, $p$. If the hoop stress is $\sigma$, the principal stresses are:

$$\sigma_1 = \sigma$$
$$\sigma_2 = \sigma/2$$
$$\sigma_3 = -p$$

According to the maximum stress theory, the controlling stress is $\sigma$, since it is the largest of the three principal stresses. According to the maximum shear stress theory, the controlling stress is the stress intensity, which is $(\sigma + p)$. Since $p$ is small in comparison with $\sigma$ for a thin-walled vessel, there is little difference between the two theories.

When a more detailed stress analysis is made, however, the difference between the two theories becomes important. A good example is the knuckle region of a dished head, where the largest stress is a meridional tension on the inside surface. This stress is accompanied by a circumferential compression, so that the stress intensity is larger than the highest stress component. For one particular case of a 2 : 1 ellipsoidal head on a 48-inch diameter vessel designed for 133 psi, the maximum stress was 23,480 psi and the highest stress intensity was 33,360 psi. The nominal hoop stress in the cylinder was 20,000 psi; thus the maximum stress theory would indicate that the localized secondary stresses only exceeded the basic design stress by 17 per cent, but the maximum shear stress theory shows that the basic design stress was exceeded by 67 per cent.

## II. STRESS CATEGORIES AND STRESS LIMITS

As mentioned previously, different types of stress require different limits, and before establishing these limits it was necessary to choose the stress categories to which limits should be applied. The categories and sub-categories chosen were as follows:

A. Primary Stress.
   (1) General primary membrane stress.
   (2) Local primary membrane stress.
   (3) Primary bending stress.
B. Secondary Stress.
C. Peak Stress.

Definitions of these terms are given in Table N-414 of Section III, but some justification for the chosen categories is in order. The major stress categories are primary, secondary, and peak. Their chief characteristics may be described briefly as follows:

(a) Primary stress is a stress developed by the imposed loading which is necessary to satisfy the laws of equilibrium between external and internal forces and moments. The basic characteristic of a primary stress is that it is not self-limiting. If a primary stress exceeds the yield strength of the material through the entire thickness, the prevention of failure is entirely dependent on the strain-hardening properties of the material.

(b) Secondary stress is a stress developed by the self-constraint of a structure. It must satisfy an imposed strain pattern rather than being in equilibrium with an external load. The basic characteristic of a secondary stress is that it is self-limiting since minor distortions can satisfy the discontinuity conditions or thermal expansions which cause the stress to occur.

(c) Peak stress is the highest stress in the region under consideration. The basic characteristic of a peak stress is that it causes no significant distortion and is objectionable mostly as a possible source of fatigue failure.

The need for dividing primary stress into membrane and bending components is obvious since, as will be shown later, limit design theory shows that the calculated value of a primary bending stress may be allowed to go higher than the calculated value of a primary membrane stress. The placing in the primary category of local membrane stress produced by mechanical loads, however, requires some explanation because this type of stress really has the basic characteristics of a secondary stress. It is self-limiting and when it exceeds yield, the external load will be resisted by other parts of the structure, but this shift may involve intolerable distortion and it was felt that it must be limited to a lower value than other secondary stresses, such as discontinuity bending stress and thermal stress.

Secondary stress could be divided into membrane and bending components, just as was done for primary stress, but after the removal of local membrane stress to the primary category, it appeared that all the remaining secondary stresses could be controlled by the same limit and this division was unnecessary.

Thermal stresses are never classed as primary stresses, but they appear in both of the other categories, secondary and peak. Thermal stresses which can produce distortion of the structure are placed in the secondary category and thermal stresses which result from almost complete suppression of the differential expansion, and thus cause no significant distortion, are classed as peak stresses.

One of the commonest types of peak stress is that produced by a notch, which might be a small hole or a fillet. The phenomenon of stress concentration is well-known and requires no further explanation here.

Many cases arise in which it is not obvious which category a stress should be placed in, and considerable judgement is required. In order to standardize this procedure and use the judgement of the writers of the Code rather than the judgement of individual designers, a table was prepared covering most of the situations which arise in pressure vessel design and specifying which category each stress must be placed in. This table appears as Table N-413 of Section III.

The grouping of the stress categories for the purpose of applying limits to the stress intensities is illustrated in Fig. N-414 of Section III. This diagram has been called the "hopper diagram" because it provides a hopper for each stress category. The calculated stresses are made to progress through the diagram in the direction of the arrows. Whenever a rectangular box appears, the sum of all the stress components which have entered the box are used to calculate the stress intensity, which is then compared to the allowable limit, shown in the circle adjacent to the rectangle. The following points should be noted in connection with this diagram:

(a) The symbols $P_m$, $P_l$, $P_b$, $Q$ and $F$ do not represent single quantities, but each represents a set of six quantities, three direct stress and three shear stress components. The addition of stresses from different categories must be performed at the component level, not after translating the stress components into a stress intensity. Similarly, the calculation of membrane stress intensity involves the averaging of stresses across a section, and this averaging must also be performed at the component level.

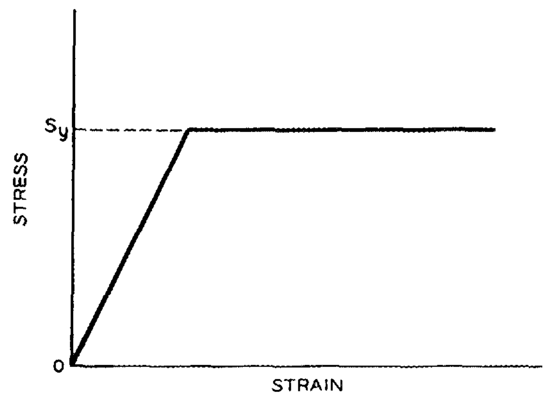
(b) The stresses in Category $Q$ are those parts of the total stress which are categorized as secondary, and do not include primary stresses which may also exist at the same point. It should be noted, however, that a detailed stress analysis frequently gives the combination of primary and secondary stresses directly, and this calculated value represents the total of $P$ (or $P_L$) + $P_b$ + $Q$ and not $Q$ alone. It is not necessary to calculate $Q$ separately since the stress limit (to be described later) applies to the total stress intensity. Similarly, if the stress in Category $F$ is produced by a stress concentration, the quantity $F$ is the additional stress produced by the notch, over and above the nominal stress, but it is not necessary to calculate $F$ separately.

**Basic Stress Intensity Limits**

The choice of the basic stress intensity limits for the stress categories described above was accomplished by the application of limit design theory tempered by some engineering judgement and some conservative simplifications. The principles of limit design which were used can be described briefly as follows.

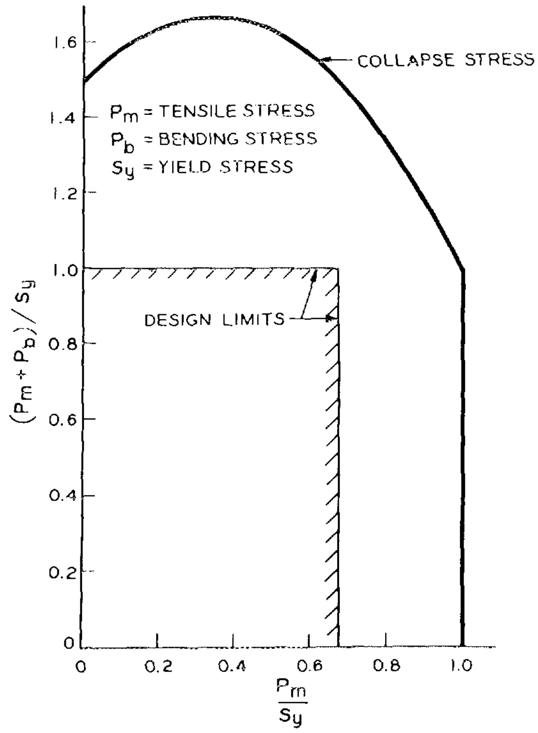
The assumption is made of perfect plasticity with no strain-hardening. This means that an idealized stress-strain curve of the type shown in Fig. 1 is assumed. Allowable stresses based on perfect plasticity and limit design theory may be considered as a floor below which a vessel made of any sufficiently ductile material will be safe. The actual strain-hardening properties of specific materials will give them larger or smaller margins above this floor.

In a structure as simple as a straight bar in tension, a load producing yield stress, $S_y$, results in "collapse". If the bar is loaded in bending, collapse does not occur until the load has been increased by a factor known as the "shape factor" of the cross section; at that time a "plastic hinge" is formed. The shape factor for a rectangular section in bending is 1.5. When the primary stress in a rectangular section consists of a combination of bending and axial tension, the value of the collapse load depends on the ratio between the tensile and bending loads. Fig. 2 shows the value of the maximum calculated stress at the outer fiber of a rectangular section which would be required to produce a plastic hinge, plotted against the average tensile stress across the section, both values expressed as multiples of the yield stress, $S_y$. When the average tensile stress, $P_m$, is zero, the failure stress for bending is 1.5 $S_y$. When the average tensile stress is $S_y$, no additional bending stress, $P_b$, may be applied.




IDEALIZED STRESS – STRAIN RELATIONSHIP

FIGURE 1.

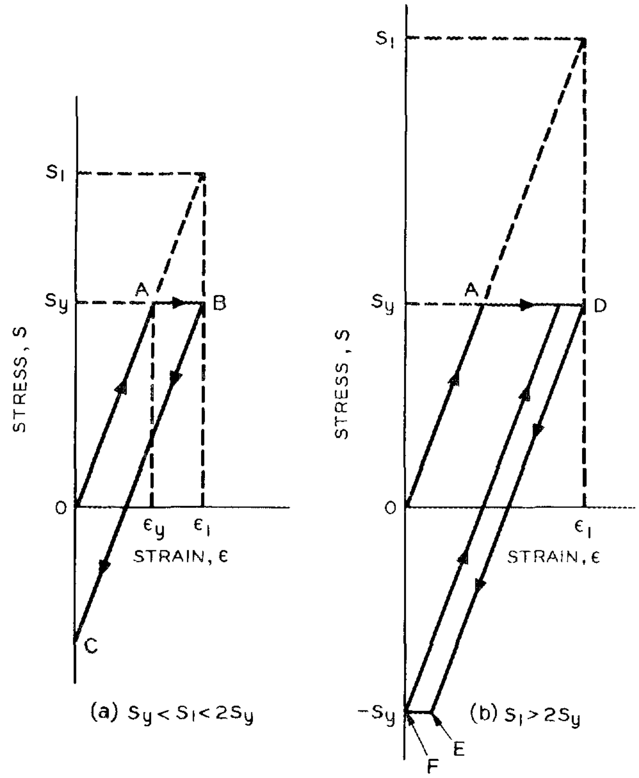



COLLAPSE STRESS FOR COMBINED TENSION AND BENDING
(RECTANGULAR SECTION)

FIGURE 2.

Figure 2 was used to choose allowable values, in terms of the yield stress, for general primary membrane stress, $P_m$, and primary membrane-plus-bending stress, $P_m + P_b$. It may be seen that limiting $P_m$ to $(2/3)S_y$ and $P_m + P_b$ to $S_y$ provides adequate safety. The safety factor is not constant for all combinations of tension and bending, but a design rule to provide a uniform safety factor would be needlessly complicated.

In the study of allowable secondary stresses, a calculated elastic stress range equal to twice the yield stress has a very special significance. It determines the borderline between loads which, when repetitively applied, allow the structure to "shake down" to elastic action and loads which produce plastic action each time they are applied. The theory of limit design provides rigorous proof of this statement, but the validity of the concept can easily be visualized. Consider, for example, the outer fiber of a beam which is strained in tension to a strain value $\epsilon_1$, somewhat beyond the yield strain as shown in Fig. 3(a) by the path $OAB$. The calculated elastic stress would be $S = S_1 = E\epsilon_1$. Since we are considering the case of a secondary stress, we shall assume that the nature of the loading is such as to cycle the strain from zero to $\epsilon_1$ and back to zero, rather than cycling the stress from zero to $S_1$, and back to zero. When the beam is returned to its undeflected position, $O$, the outer fiber has a residual compressive stress of magnitude $S_1 - S_y$. On any subsequent loading, this residual compression must be removed before the stress goes into tension and thus the elastic range has been increased by the quantity $S_1 - S_y$. If $S_1 = 2S_y$, the elastic range becomes $2S_y$, but if $S_1 > 2S_y$, the fiber yields in compression, as shown by $EF$ in Fig. 3(b) and all subsequent cycles produce plastic strain. Therefore, $2S_y$ is the maximum value of calculated secondary elastic stress which will "shake down" to purely elastic action.




STRAIN HISTORY BEYOND YIELD

FIGURE 3.

An important point to note from the foregoing discussion of primary and secondary stresses is that 1.5 $S_y$ is the *failure* stress for primary bending, whereas for secondary bending $2S_y$ is merely the threshold beyond which some plastic action occurs. Therefore the allowable design stress for primary bending must be reduced below $1.5S_y$ to, say, $1.0S_y$, whereas $2S_y$ is a safe design value for secondary bending since a little plastic action during overloads is tolerable. The same type of analysis shows that $2S_y$ is also a safe design value for secondary membrane tension. As described previously, local membrane stress produced by mechanical load has the characteristics of a secondary stress but has been arbitrarily placed in the primary category. In order to avoid excessive distortion, it has been assigned an allowable stress level of $S_y$, which is 50 per cent higher than the allowable for general primary membrane stress but precludes excessive yielding.

We have now shown how the allowable stresses for the first four stress categories listed in the previous section should be related to the yield strength of the material. The last category, peak stress, is related only to fatigue, and will be discussed later. In Section III the allowables are not expressed in terms of the yield strength, but rather as multiples of the tabulated value $S_m$, which is the allowable for general primary membrane stress. In assigning allowable stress values to a variety of materials with widely varying ductilities and widely varying strain-hardening properties, the yield strength alone is not a sufficient criterion. In order to prevent unsafe designs in materials with low ductility and in materials with high yield-to-tensile ratios, the Code has always considered both the yield strength and the ultimate tensile strength in assigning allowable stresses. This principle has not been changed in Section III but the chosen fractions of the mechanical properties have been increased to two-thirds yield strength and one-third ultimate strength instead of five-eighths yield strength (for ferrous materials) and one-fourth ultimate strength. The Special Committee believed that this increase was quite safe because the detailed stress analysis required by Section III eliminates the need for a large safety factor to cover unanalyzed areas. The stress intensity limits for the various categories given in Section III are such that the multiples of yield strength described above are never exceeded. Table I summarizes the basic stress limits of Section III and shows the multiples of yield strength and ultimate strength which these limits do not exceed.

TABLE I

BASIC STRESS INTENSITY LIMITS

| Categories | Stress Intensity Limit in Terms of: Tabulated $S_m$ value | Yield Strength ($S_y$) | Ultimate Tensile Strength ($S_u$) |
|---|---|---|---|
| General primary membrane stress intensity ($P_m$) | $S_m$ | $\leqslant \frac{2}{3} S_y$ | $\leqslant \frac{1}{3} S_u$ |
| Local primary membrane stress intensity ($P_l$) | $1.5\ S_m$ | $\leqslant S_y$ | $\leqslant \frac{1}{2} S_u$ |
| Primary membrane plus bending stress intensity ($P_m$ (or $P_l$) + $P_b$) | $1.5\ S_m$ | $\leqslant S_y$ | $\leqslant \frac{1}{2} S_u$ |
| Primary plus secondary stress intensity ($P_m$ (or $P_l$) + $P_b$ + $Q$ | $3\ S_m$ | $\leqslant 2\ S_y$ | $\leqslant S_u$ |

## Stresses Above Yield Strength

The primary criterion of the structural adequacy of a design, according to the rules of Section III, is that the stresses, as determined by calculation or experimental stress analysis, shall not exceed the specified allowable limits. It frequently happens that both the calculated stress and the allowable stress exceeds the yield strength of the material. Nevertheless, unless stated specifically otherwise, it is expected that calculations be made on the assumption of elastic behavior.

Allowable stresses higher than yield appear in the values for primary-plus-secondary stress and in the fatigue curves. In the case of the former, the justification for allowing calculated stresses higher than yield is that the limits are such as to assure shake-down to elastic action after repeated loading has established a favorable pattern of residual stresses. Therefore the assumption of elastic behavior is justified because it really exists in all load cycles subsequent to shake-down.

In the case of fatigue analysis, plastic action can actually persist throughout the life of the vessel, and the justification for the specified procedure is somewhat different. Repetitive plastic action occurs only as the result of peak stresses in relatively localized regions and these regions are intimately connected to larger regions of the vessel which behave elastically. A typical example is the peak stress at the root of a notch, in a fillet, or at the edge of a small hole. The material in these small regions is strain-cycled rather than stress-cycled (as will be discussed later) and the elastic calculations give numbers which have the dimensions of stress but are really proportional to the strain. The factor of proportionality for uniaxial stress is, of course, the modulus of elasticity. The fatigue curves in Section III have been specially designed to give numbers comparable to these fictitious calculated stresses. The curves are based on strain-cycling data, and the strain values have been multiplied by the modulus of elasticity. Therefore stress intensities calculated from the familiar formulas of strength-of-materials texts are directly comparable to the allowable stress values in the fatigue curves.

## III. FATIGUE ANALYSIS

One of the important innovations in Section III, as compared to Sections I and VIII, is the recognition of fatigue as a possible mode of failure and the provision of specific rules for its prevention. Fatigue has been a major consideration for many years in the design of rotating machinery and aircraft, where the expected number of cycles is in the millions and can usually be considered infinite for all practical purposes. For the case of large numbers of cycles, the primary concern is the endurance limit, which is the stress which can be applied an infinite number of times without producing failure. In pressure vessels, however, the number of stress cycles applied during the specified life seldom exceeds $10^5$ and is frequently only a few thousand. Therefore, in order to make fatigue analysis practical for pressure vessels, it was necessary to develop some new concepts not previously used in machine design [1, 2].

In Section III there are some differences between the procedures for fatigue analysis which are specified for materials with tensile strengths below 100,000 psi and those specified for materials with tensile strengths of 100,000 psi and greater. In vessels covered by Section III, the latter materials appear only in high-strength bolting and one seamless heat-treated forging on which no welding is permitted after heat treatment. In the course of studying the fatigue problem, the Special Committee first developed a set of procedures applicable to all ductile materials and found later that for the lower-strength types, with the lower yield-to-tensile ratios, some conservative simplifications of these procedures were both feasible and advisable. Thus the complete form of the procedure is only used for high-strength bolting and most components are evaluated by the simpler method. In order to understand the simpler method, however, it is first necessary to understand the complete method. Therefore the following discussion will first describe the complete procedure and then show how it was simplified to the form recommended in Section III, Par. N-415.

### Use of Strain-Controlled Fatigue Data

The chief difference between high-cycle fatigue and low-cycle fatigue is the fact that the former involves little or no plastic action, whereas failure in a few thousand cycles can be produced only by strains in excess of the yield strain. In the plastic region large changes in strain can be produced by small changes in stress. Fatigue damage in the plastic region has been found to be a function of plastic strain and therefore fatigue curves for use in this region should be based on tests in which strain rather than stress is the controlled variable. As a matter of convenience, the strain values used in the tests are multiplied by the elastic modulus to give a fictitious stress which is not the actual stress applied but has the advantage of being directly comparable to stresses calculated on the assumption of elastic behavior.

The use of strain instead of stress and the consideration of plastic action have necessitated some additional departures from the conventional methods of studying fatigue problems. It has been common practice in the past to use lower stress concentration factors for small numbers of cycles than for large numbers of cycles. This is reasonable when the allowable stresses are based on stress-fatigue data, but is not advisable when strain-fatigue data are used. Fig. 4 shows typical relationships between stress, $S$, and cycles-to-failure, $N$, from (A) strain cycling tests on unnotched specimens, (B) stress-cycling tests on unnotched specimens, and (C) stress-cycling tests on notched specimens. The ratio between the ordinates of curves (B) and (C) decreases with decreasing cycles-to-failure, and this is the basis for the commonly-accepted practice of using lower values

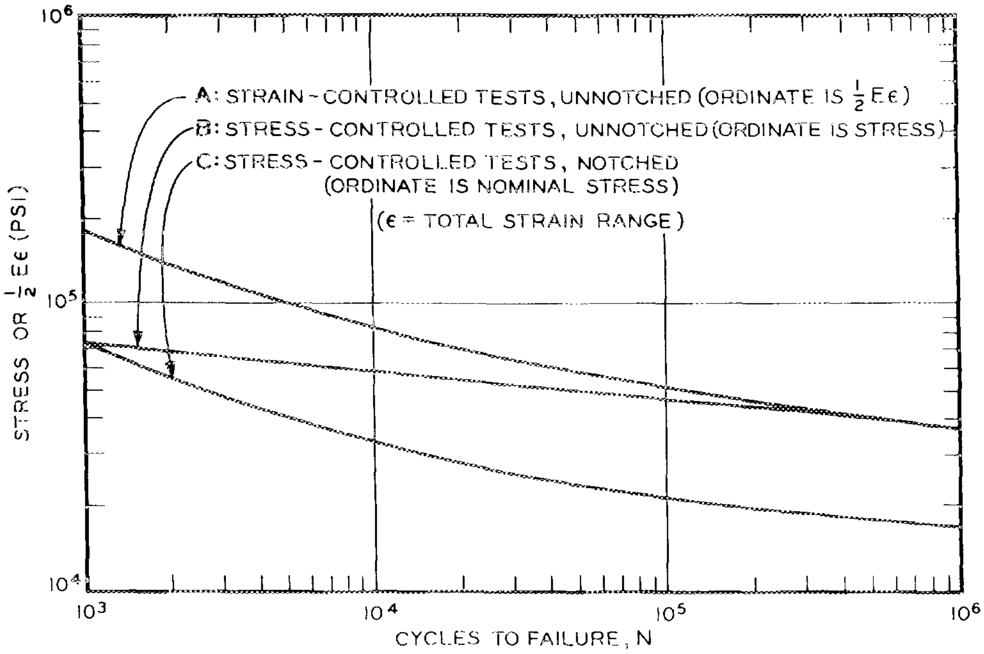



FIG. 4. TYPICAL RELATIONSHIP BETWEEN STRESS, STRAIN, AND CYCLES-TO-FAILURE.

of $K$ (stress concentration factor) for lower values of $N$. In (C), however, although nominal stress is the controlled parameter, the material in the root of the notch is really being strain cycled, because the surrounding material is at a lower stress and behaves elastically. Therefore it should be expected that the ratio between curves (A) and (C) should be independent of $N$ and equal to $K$. For this reason it is recommended in Section III that the same value of $K$ be used regardless of the number of cycles involved.

The choice of an appropriate stress concentration factor is not an easy one to make and Section III gives some guidance in this area. For fillets, grooves, holes, etc. of known geometry, it is safe to use the theoretical stress concentration factors found in such references as [3] and [4], even though strain concentrations can sometimes exceed the theoretical stress concentration factors. The use of the theoretical factor as a safe upper limit is justified, however, since strain concentrations higher than the stress concentrations only occur when gross yielding is present in the surrounding material, and this situation is prevented by the use of basic stress limits which assure shake-down to elastic action. For very sharp notches it is well known that the theoretical factors grossly overestimate the true weakening effect of the notch in the low and medium strength materials used for pressure vessels. Therefore no factor higher than 5 need ever be used for any configuration allowed by the design rules and an upper limit of 4 is specified for some specific constructions such as fillet welds and screw threads. When fatigue tests are made to find the appropriate factor for a given material and configuration, they should be made with a material of comparable notch sensitivity and failure should occur in a reasonably large number of cycles (> 1000) so that the test does not involve gross yielding.

### Effect of Mean Stress

Another deviation from common practice occurs in the consideration of fluctuating stress, which is a situation where the stress fluctuates around a mean value different from zero, as shown in Fig. 5. The evaluation of the effects of mean stress is commonly accomplished by use of the modified Goodman diagram, as shown in Fig. 6, where mean

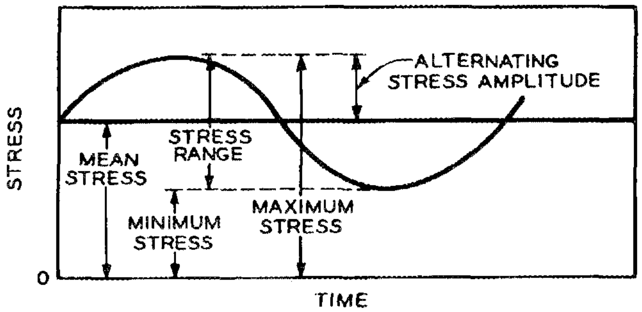



FIG. 5. STRESS FLUCTUATION AROUND A MEAN VALUE.

stress is plotted as the abscissa and the amplitude (half range) of the fluctuation is plotted as the ordinate. The straight line joining the endurance limit, $S_e$, (where $S_N = S_e$) on the vertical axis (point $E$) with the ultimate strength, $S_u$, on the horizontal axis (point $D$) is a conservative approximation of the combinations of mean and alternating stress which produce failure in large numbers of cycles. A little consideration of this diagram shows that not all points below the "failure" line, $ED$, are feasible. Any combination of mean and alternating stresses which results in a stress excursion above the yield strength will produce a shift in the mean stress which keeps the maximum stress during the cycle at the yield value. This shift has already been illustrated by the strain history shown in Fig. 3. The feasible combinations of mean and alternating stress are all contained within the 45 degree triangle $AOB$ or on the vertical axis above $A$, where $A$ is the yield strength on the vertical axis and $B$ is the yield strength on the horizontal axis. Regardless of the conditions under which any test or service cycle is started, the true conditions after the application of a few cycles must fall within this region because all combinations above $AB$ have a maximum stress above yield and there is a consequent reduction of mean stress which shifts the conditions to a point on the line $AB$ or all the way to the vertical axis.

It may be seen from the foregoing discussion that the value of mean stress to be used in the fatigue evaluation is not always the value which is calculated directly from the imposed loading cycle. When the loading cycle produces calculated stresses which exceed

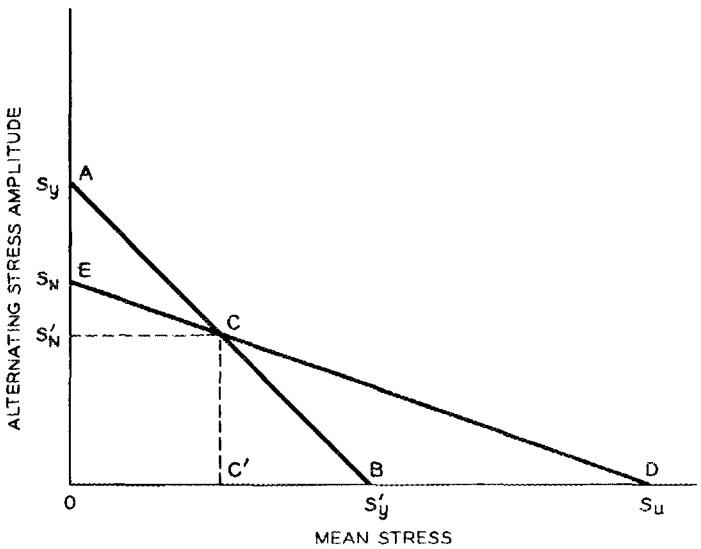



FIG. 6. MODIFIED GOODMAN DIAGRAM.

the yield strength at any time, it is necessary to calculate an adjusted value of mean stress before completing the fatigue evaluation. The rules for calculating this adjusted value may be summarized as follows.

Let $S'_{mean}$ = basic value of mean stress (calculated directly from loading cycle)

$S_{mean}$ = adjusted value of mean stress

$S_{alt}$ = amplitude (half range) of stress fluctuation

$S_y$ = yield strength

$$\left.\begin{array}{l} \text{If } S_{alt} + S'_{mean} \leqslant S_y,\ S_{mean} = S'_{mean} \\ \text{If } S_{alt} + S'_{mean} > S_y \text{ and } S_{alt} < S_y,\ S_{mean} = S_y - S_{alt} \\ \text{If } S_{alt} \geqslant S_y,\ S_{mean} = 0. \end{array}\right\} \quad (2)$$

The fatigue curves are based on tests involving complete stress reversal, that is, $S_{mean} = 0$. Since the presence of a mean stress component detracts from the fatigue resistance of the material, it is necessary to determine the equivalent alternating stress component for zero mean stress before entering the fatigue curve. This quantity, designated $S_{eq}$, is the alternating stress component which produces the same fatigue damage at zero mean stress as the actual alternating stress component, $S_{alt}$, produces at the existing value of mean stress. It can be obtained graphically from the Goodman diagram by projecting a line as shown in Fig. 7 from $S_u$ through the point $(S_{mean}, S_{alt})$ to the vertical axis. It is usually easier, however, to use the simple formula

$$S_{eq} = \frac{S_{alt}}{1 - \dfrac{S_{mean}}{S_u}} \quad (3)$$

$S_{eq}$ is the value of stress to be used in entering the fatigue curve to find the allowable number of cycles.

The foregoing discussion of mean stress and the shift which it undergoes when yielding occurs leads to another necessary deviation from standard procedures which is used in Section III. In applying stress concentration factors to the case of fluctuating stress, it has been the common practice to apply the factor to only the alternating component. This is not a logical procedure, however, because the material will respond in the same way to a given load regardless of whether the load will later turn out to be steady or fluctuating. It is more logical to apply the concentration factor to both the mean and the alter-

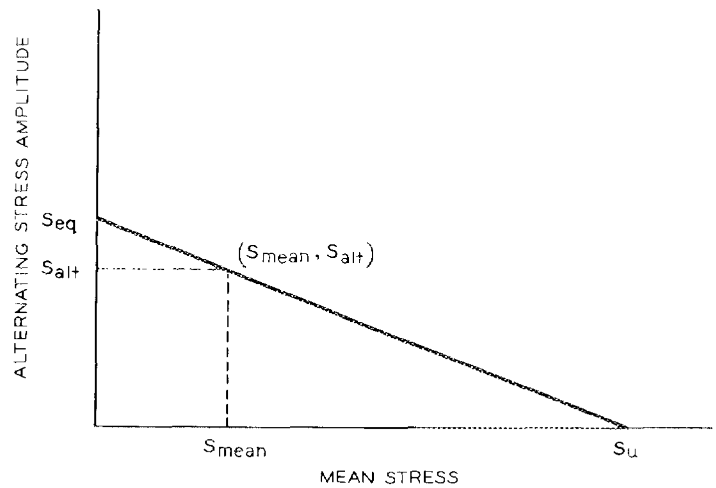
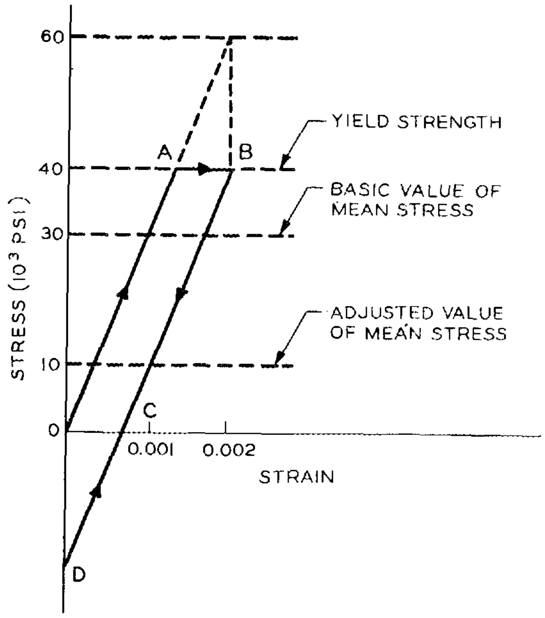



FIG. 7. GRAPHICAL DETERMINATION OF $S_{eq}$.




IDEALIZED STRESS VS STRAIN HISTORY
FIGURE 8.

nating component and then consider the reduction which yielding produces in the mean component. It is important to remember that the concentration factor must be applied before the adjustment for yielding is made. The following example shows that the common practice of applying the concentration factor to only the alternating component gives a rough approximation to the real situation but can sometimes be unconservative.

Take the case of a material with 80,000 psi tensile strength, 40,000 psi yield strength and $30 \times 10^6$ psi modulus made into a notched bar with a stress concentration factor of 3. The bar is cycled between nominal tensile stress values of 0 and 20,000 psi. Common practice would call $S_{mean}$, the mean stress, 10,000 psi and $S_{alt}$, the alternating component, $(1/2) \times 3 \times 20{,}000 = 30{,}000$ psi. The stress-strain history of the material at the root of the notch would be, in idealized form, as shown in Fig. 8. The calculated maximum stress, assuming elastic behavior, is 60,000 psi. The basic value of mean stress, $S'_{mean}$, is 30,000 psi, but since $S_{alt} + S'_{mean} = 60{,}000 \text{ psi} > S_y$ and $S_{alt} = 30{,}000 \text{ psi} < S_y$,

$$S_{mean} = S_y - S_{alt} = 40{,}000 - 30{,}000 = 10{,}000 \text{ psi}$$

and

$$S_{eq} = \frac{30{,}000}{1 - \dfrac{10{,}000}{80{,}000}} = 34{,}300 \text{ psi.}$$

It so happens that, for the case chosen, the common practice gives exactly the same result as the proposed method. Thus, the yielding during the first cycle is seen to be the justification for the common practice of ignoring the stress concentration factor when determining the mean stress component. The common practice, however, would have given the same result regardless of the yield strength of the material, whereas the proposed method gives different mean stresses for different yield strengths. For example, if the yield strength had been 50,000 psi, $S_{mean}$ would have been 20,000 psi and $S_{eq}$ would have been 40,000 psi. The common practice would have given 34,300 psi, an unconservative result.

As was mentioned previously, the complete procedure for fatigue analysis, as outlined above, is used in Section III only for the case of high-strength bolting. For other parts of the structure, particularly if welding is used, the residual stress may produce a value of mean stress higher than that calculated by the procedure. Therefore it would be advisable and also much easier to adjust the fatigue curve downward enough to allow for the maximum possible effect of mean stress. It will be shown here that this adjustment is small for the case of low and medium-strength materials but that it is unduly conservative for materials with high yield-to-tensile ratios.

As a first step in finding the required adjustment of the fatigue curve, let us find how the mean stress affects the amplitude of alternating stress which is required to produce fatigue failure. In the modified Goodman diagram of Fig. 6 it may be seen that at zero mean stress the required amplitude for failure in $N$ cycles is designated $S_N$. As the mean stress increases along $OC'$, the required amplitude of alternating stress decreases along the line $EC$. If we try to increase the mean stress beyond $C'$, yielding occurs and the mean stress reverts to $C'$. Therefore $C'$ represents the highest value of mean stress which has any effect on fatigue life. Since $S_N'$ in Fig. 6 is the alternating stress required to produce failure in $N$ cycles when the mean stress is at $C'$, $S_N'$ is the value to which the point on the fatigue curve at $N$ cycles must be adjusted if the effects of mean stress are to be ignored. From the geometry of Fig. 6, it can be shown that

$$S_N' = S_n \left[ \frac{S_u - S_y}{S_u - S_N} \right] \quad \text{for } S_N < S_y \tag{4}$$

When $N$ decreases to the point where $S_N \geqslant S_y$, then $S_N' = S_N$ and no adjustment of this region of the curve is required.

Figures 9, 10 and 11 show the fatigue data which were used to construct the design fatigue curves of Section III. In each case the solid line is the best-fit failure curve for zero mean stress and the dotted line is the curve adjusted in accordance with (4). Fig. 11 for stainless steel and nickel-chrome-iron alloy has no dotted line because the fatigue limit is higher than the yield strength over the whole range of cycles. In Section III a single design curve is used for carbon and low-alloy steels because, as may be noted from Figs. 9 and 10, the adjusted curves for these classes of material were nearly identical.

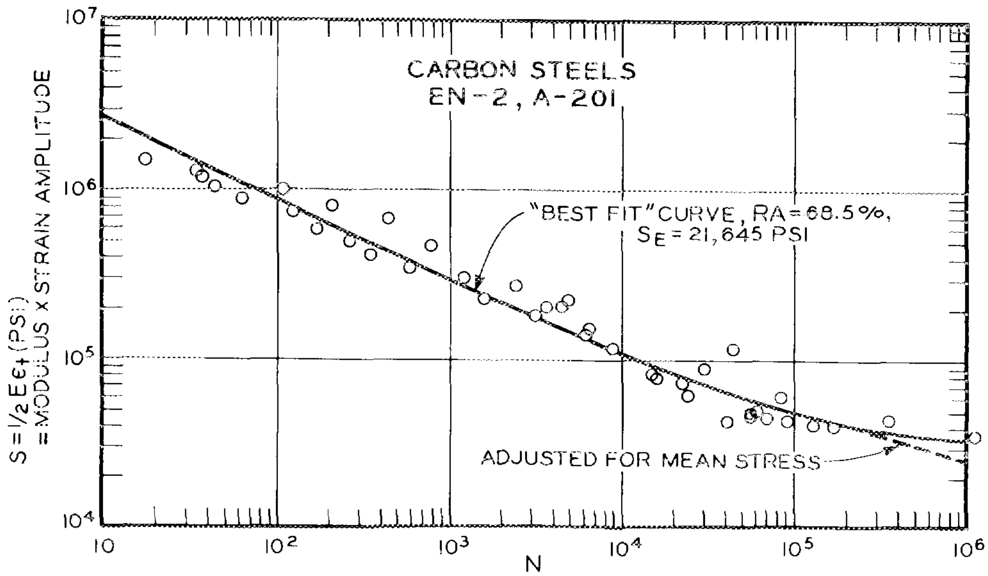



FIG. 9. FATIGUE DATA - CARBON STEELS.

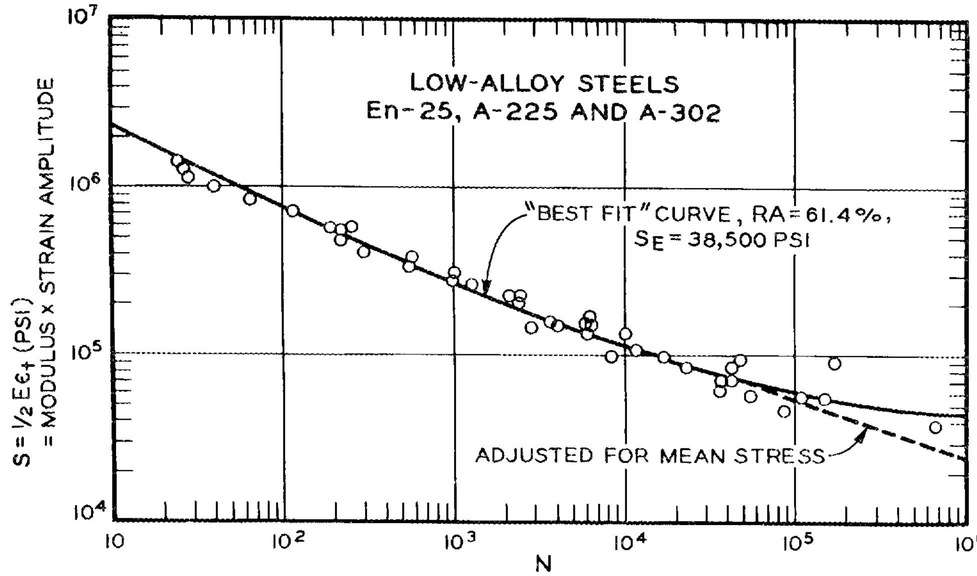



FIG. 10. FATIGUE DATA – LOW-ALLOY STEELS.

For the case of high-strength, heat-treated, bolting materials, the heat treatment increases the yield strength of the material much more than it increases either the ultimate strength, $S_u$, or the fatigue limit, $S_N$. Inspection of (4) shows that for such cases, $S_N'$ becomes a small fraction of $S_N$ and thus the correction for the maximum effect of mean stress becomes unduly conservative. Furthermore, bolts are not apt to have uncontrolled residual

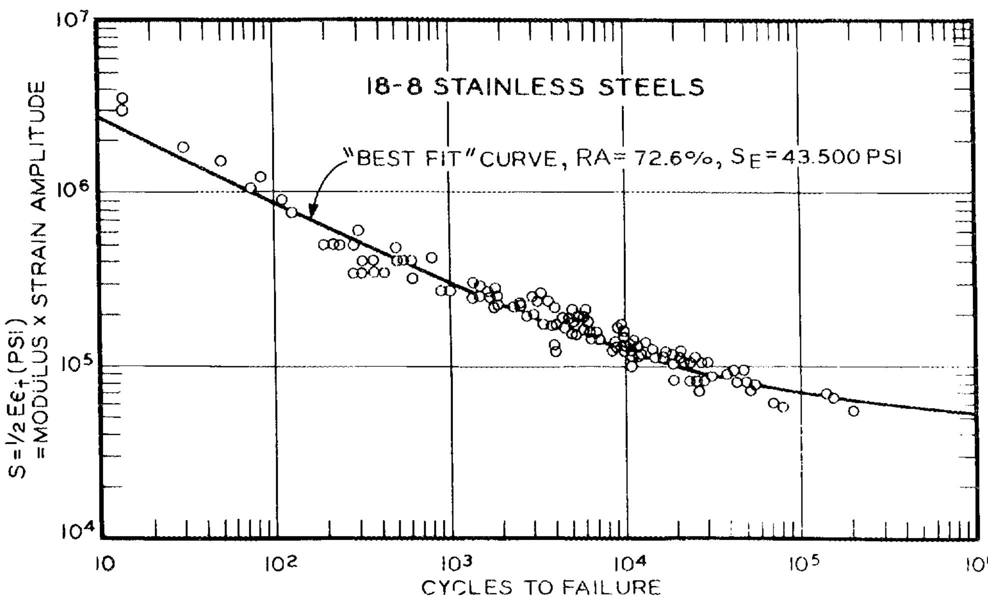



FIG. 11. FATIGUE DATA – STAINLESS STEELS.

stresses such as are produced by casting or welding processes, and thus the extra conservatism is not warranted.

For the fatigue analysis of high-strength bolts the design fatigue curve is replaced by the analytical approximation

$$N = \left[\frac{10^6 + 0.64 S_u}{S_{eq} - 0.2\, S_u}\right]^2 \tag{5}$$

where $S_u$ = specified minimum tensile strength (psi)

and $$S_{eq} = \frac{S_{alt}}{1 - \dfrac{S_{mean}}{S_u}}$$

This expression represents a fatigue curve which has the same safety factors as the fatigue curves for lower strength material (to be given later) but which has not been corrected for the maximum effect of mean stress. Thus the effects of the mean stress produced by the applied loading cycle must be considered. The expression was derived from the following assumptions, all of which have been justified by a considerable amount of experimental data:

(1) The relationship between $S$ (strain amplitude times elastic modulus) and $N$ (cycles-to-failure) can be represented as a function of the ductility, the modulus, and the endurance limit as follows [2]:

$$S = \frac{E}{4\sqrt{N}} \ln \frac{100}{100 - RA} + S_e \tag{6}$$

where $E$ = elastic modulus(psi)
$RA$ = per cent reduction of area in tensile test
$S_e$ = endurance limit (psi)

(2) The minimum reduction of area of the bolting materials covered is 45 per cent.

(3) The endurance limit of the bolting materials covered is approximately 40 per cent of the ultimate tensile strength.

**Procedure for Fatigue Evaluation**

The step-by-step procedure for determining whether or not the fluctuation of stresses at a given point is acceptable is given in detail in Par. N-415.2 of Section III. The procedure is based on the maximum shear stress theory of failure and consists of finding the amplitude (half full range) through which the maximum shear stress fluctuates. Just as in the case of the basic stress limits, the stress differences and stress intensities (twice maximum shear stress) are used in place of the shear stress itself.

At each point on the vessel at any given time there are three principal stresses, $\sigma_1$, $\sigma_2$, and $\sigma_3$ and three stress differences, $S_{12}$, $S_{23}$, and $S_{31}$. The stress intensity is the largest of the three stress differences and is usually considered to have no direction or sign, just as for the strain energy of distortion. When considering fluctuating stresses, however, this concept of non-directionality can lead to errors when the sign of the shear stress changes during the cycle. Therefore the range of fluctuation must be determined from the stress differences in order to find the full algebraic range. The alternating stress intensity, $S_{alt}$, is the largest of the amplitudes of the three stress differences. This feature of being able to maintain directionality and thus find the algebraic range of fluctuation is one reason why the maximum shear stress theory rather than the strain energy of distortion theory was chosen for Section III.

When the directions of the principal stresses change during the cycle (regardless of whether the stress differences change sign), the non-directional strain energy of distortion theory breaks down completely. This has been demonstrated experimentally by Findley and his associates [5] who produced fatigue failures in a rotating specimen compressed across a diameter. The load was fixed while the specimen rotated. Thus the principal stresses rotated but the strain energy of distortion remained constant. The procedure outlined in Par. N-415.2(b) is consistent with the results of Findley's tests and uses the range of shear stress on a fixed plane as the criterion of failure. The procedure brings in the effect of rotation of the principal stresses by considering only the *changes* in shear stress which occur in each plane between the two extremes of the stress cycle.

### Cumulative Damage

In many cases a point on a vessel will be subjected to a variety of stress cycles during its lifetime. Some of these cycles will have amplitudes below the endurance limit of the material and some will have amplitudes of varying amounts above the endurance limit. The cumulative effect of these various cycles is evaluated in Section III by means of a linear damage relationship in which it is assumed that if $N_1$ cycles would produce failure at a stress level $S_1$, then $n_1$ cycles at the same stress level would use up the fraction $n_1/N_1$ of the total life. Failure occurs when the cumulative usage factor, which is the sum $n_1/N_1 + n_2/N_2 + n_3/N_3 + \dots$ is equal to 1.0. Other hypotheses for estimating cumulative fatigue damage have been proposed and some have been shown to be more accurate than the linear damage assumption. Better accuracy could be obtained, however, only if the sequence of the stress cycles were known in considerable detail, and this information is not apt to be known with any certainty at the time the vessel is being designed. Tests have shown [6] that the linear assumption is quite good when cycles of large and small stress magnitude are fairly evenly distributed throughout the life of the member, and therefore this assumption was considered to cover the majority of cases with sufficient accuracy. It is of interest to note that a concentration of the larger stress cycles near the beginning of life tends to accelerate failure, whereas if the smaller stresses are applied first and followed by progressively higher stresses, the cumulative usage factor can be "coaxed" up to a value as high as 4 or 5.

When stress cycles of various frequencies are intermixed through the life of the vessel, it is important to identify correctly the range and number of repetitions of each type of cycle. It must be remembered that a small increase in stress range can produce a large decrease in fatigue life, and this relationship varies for different portions of the fatigue curve. Therefore the effect of superposing two stress amplitudes cannot be evaluated by adding the usage factors obtained from each amplitude by itself. The stresses must be added before calculating the usage factors. Consider, for example, the case of a thermal transient which occurs in a pressurized vessel. Suppose that at a given point the pressure stress is 20,000 psi tension and the added stress from the thermal transient is 70,000 psi tension. If the thermal cycle occurs 10,000 times during the design life and the vessel is pressurized 1000 times, the usage factor should be based on 1000 cycles with a range from zero to 90,000 psi and 9000 cycles with a range from 20,000 psi to 90,000 psi. Another example, is given in N-415.2(d)(1).

### Exemption from Fatigue Analysis

The fatigue analysis of a vessel is quite apt to be one of the most laborious and time-consuming parts of the design procedure and this engineering effort is not warranted for vessels which are not subjected to cyclic operation. However, there is no obvious borderline between cyclic and non-cyclic operation. No operation is completely non-cyclic, since startup and shutdown is itself a cycle. Therefore, fatigue cannot be completely ignored, but Par. N-415.1 gives a set of rules which may be used to justify the by-passing of the detailed fatigue analysis for vessels in which the danger of fatigue failure is remote. The application of these rules requires only that the designer know the specified pressure fluctuations and that he have some knowledge of the temperature differences which will exist between different points in the vessel. He does not need to determine stress concentration factors or to calculate cyclic thermal stress ranges. He must, however, be sure that the basic stress limits of N-414.1 to N-414.4 are met, which may involve some calculation of the most severe thermal stresses.

The rules of N-415.1 are based on a set of assumptions, some of which are highly conservative and some of which are not conservative, but it is believed that the conservatisms outweigh the unconservatisms. These assumptions are:

(1) The worst geometrical stress concentration factor to be considered is 2. This assumption is unconservative since $K = 4$ is specified for some geometries.

(2) The concentration factor of 2 occurs at a point where the nominal stress is $3\,S_m$, the highest allowable value of primary-plus-secondary stress. This is a conservative assumption. The net result of assumptions 1 and 2 is that the peak stress due to pressure is assumed to be $6\,S_m$, which appears to be a safe assumption for a good design.

(3) *All* significant pressure cycles and thermal cycles have the same stress range as the *most severe* cycle. This is a highly conservative assumption. (A "significant" cycle is defined as one which produces a stress amplitude higher than the endurance limit of the material).

(4) The highest stress produced by a pressure cycle does not coincide with the highest stress produced by a thermal cycle. This is unconservative and must be balanced against the conservatism of assumption 3.

(5) The calculated stress produced by a temperature difference $\Delta T$ between two points does not exceed $2E\alpha\Delta T$, but the peak stress is raised to $4E\alpha\Delta T$ because of the assumption that a $K$ value of 2 is present. This assumption is conservative, as evidenced by the following examples of thermal stress:

(a) For the case of a linear thermal gradient through the thickness of a vessel wall, if the temperature difference between the inside and the outside of the wall is $\Delta T$, the stress is

$$\sigma = \frac{E\alpha\Delta T}{2(1-\nu)} = .715\,E\alpha\Delta T \text{ (for } \nu = 0.3) .$$

(b) When a vessel wall is subjected to a sudden change of temperature, $\Delta T$, so that the temperature change only penetrates a short distance into the wall thickness, the thermal stress is

$$\sigma = \frac{E\alpha\Delta T}{1-\nu} = 1.43\,E\alpha\Delta T \text{ (for } \nu = 0.3) .$$

(c) When the average temperature of a nozzle is $\Delta T$ degrees different from that of the rigid wall to which it is attached, the upper limit to the magnitude of the discontinuity stress is

$$\sigma = 1.83\,E\alpha\Delta T \text{ (for } \nu = 0.3).$$

Thus the coefficient of $E\alpha\Delta T$ is always less than the assumed value of 2.0.

When the two points in the vessel whose temperatures differ by $\Delta T$ are separated from each other by more than $2\sqrt{Rt}$, there is sufficient flexibility between the two points to produce a significant reduction in thermal stress. Therefore only temperature differences between "adjacent" points need be considered.

### Experimental Verification of Design Fatigue Curves

The design fatigue curves of Figs. N-415 (a) and (b) are based primarily on strain-controlled fatigue tests of small polished specimens. A best-fit to the experimental data

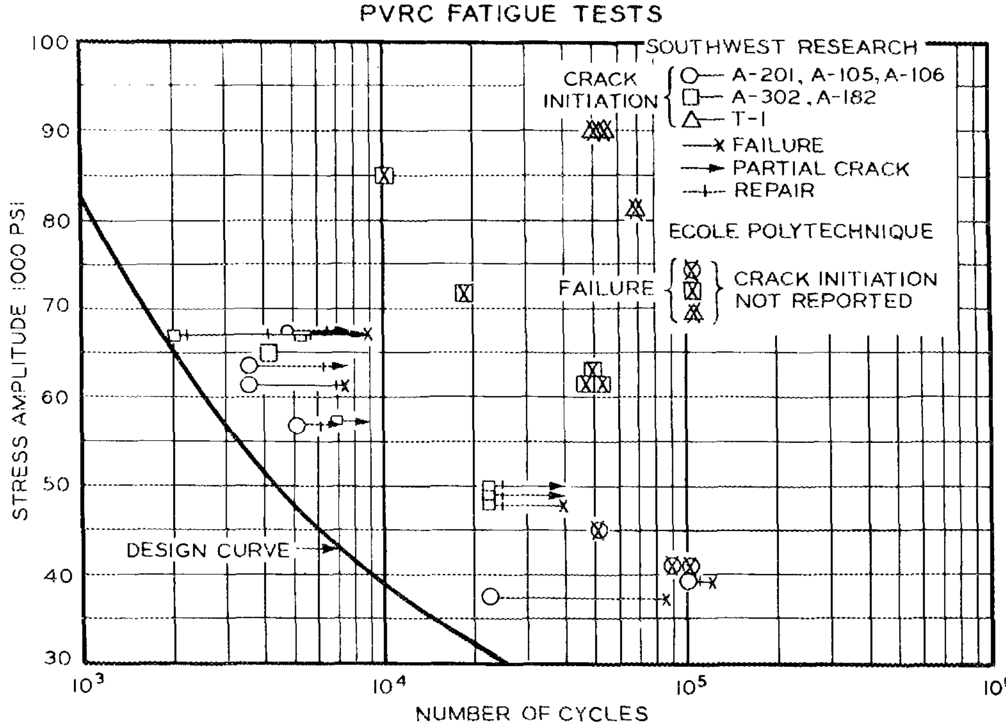



FIG. 12. PVRC FATIGUE TESTS.

was obtained by applying the method of least squares to the logarithms of the experimental values. The design stress values were obtained from the best-fit curves by applying a factor of two on stress or a factor of twenty on cycles, whichever was more conservative at each point. These factors were intended to cover such effects as environment, size effect, and scatter of data, and thus it is not to be expected that a vessel will actually operate safely for twenty times its specified life.

The appropriateness of the chosen safety factors for fatigue has recently been demonstrated by tests conducted by the Pressure Vessel Research Committee [7, 8]. In these tests 12-inch diameter model vessels and 3-foot diameter full-size vessels were tested by cyclic pressurization after a comprehensive strain gage survey was made of the peak stresses. Fig. 12 shows a summary of the PVRC test results compared to the recommended design fatigue curve of Section III for carbon and low-alloy steel. It may be seen that no crack initiation was detected at any stress level below the allowable stress, and no crack progressed through a vessel wall in less than three times the allowable number of cycles. The large scatter of the data does indicate that further research on specific materials and further studies of nozzle stresses could eventually lead to less restrictive rules for some materials and some nozzle designs.

## IV. CREEP AND STRESS-RUPTURE

It is an observed characteristic of pressure vessel materials that in service above a certain temperature, which varies with the alloy composition, the materials undergo a continuing deformation (creep) at a rate which is strongly influenced by both stress and temperature. In order to prevent excessive deformation and possible premature rupture it is necessary to limit the allowable stresses by additional criteria on creep-rate and stress-rupture. In this creep range of temperatures these criteria may limit the allowable stress to substantially lower values than those suggested by the usual factors on short time tensile and yield strengths. Satisfactory empirical limits for creep-rate and stress-rupture have been established and used in Section I and Section VIII.

Creep behavior complicates the detailed stress analysis, which is the basis of Section III, because the distribution of stress will vary with time as well as with the applied loads. The difficulties are particularly noticeable under cyclic loading. It has not yet been possible for the Special Committee to formulate complete design criteria and rules for Section III in the creep range, and the present application of Section III is restricted to temperatures at which creep will not be significant. This has been done by limiting the tabulated allowable stress intensities to below the temperature of creep behavior.

## V. SUMMARY

The design criteria of Section III differ from those of Sections I and VIII in the following respects:

(a) Section III uses the maximum shear stress (Tresca) theory of failure instead of the maximum stress theory.

(b) Section III requires the detailed calculation and classification of all stresses and the application of different stress limits to different classes of stress, whereas Sections I and VIII give formulas for minimum allowable wall thickness.

(c) Section III requires the calculation of thermal stresses and gives allowable values for them, whereas Sections I and VIII do not.

(d) Section III considers the possibility of fatigue failure and gives rules for its prevention, whereas Sections I and VIII do not.

The stress limits of Section III are intended to prevent three different types of failure, as follows:

(a) Bursting and gross distortion from a single application of pressure are prevented by the limits placed on primary stresses.

(b) Progressive distortion is prevented by the limits placed on primary-plus-secondary stresses. These limits assure shake-down to elastic action after a few repetitions of the loading.

(c) Fatigue failure is prevented by the limits placed on peak stresses.

The design criteria described here were developed by the joint efforts of the members of the Special Committee to Review the Code Stress Basis and its Task Groups over a period of several years. It is not to be expected that this paper will answer all the questions which will be asked, but it is hoped that it will give sufficient background to justify the rules which have been given.

## FABRICATION AND INSPECTION

The rules of this Section of the Code were divided to cover the various classes of vessels that might be required in a nuclear installation. The rules covering fabrication and inspection of Class A vessels for reactor or primary service were developed for a single quality, including 100 per cent radiography of all butt welds, keeping in mind that periodic inspection or maintenance might be difficult or even impossible.

The applicable rules from Section VIII for pressure vessels in lethal service were tightened where it was felt necessary, and the special requirements of the Code Cases for nuclear vessels were incorporated. Some of the special requirements are:

(a) Emphasis has been placed upon repairs during construction.

(b) Specific requirements have been spelled out, including material and welding requirements for attachments to pressure parts.

(c) The tolerance rules in Section VIII for external pressure vessels were extended to cover spherical and conical shells as well as cylindrical shells and were made applicable to internal pressure vessels as well as external.

(d) Rules covering methods and evaluation for nondestructive testing other than radiography include magnetic particle examination, liquid penetrant examination and ultrasonic examination. The manufacturer must certify as to the qualifications of his operators who conduct these examinations and the authorized inspector must satisfy himself that proper personnel and procedures are used.

(e) Post-weld heat treatment of completed vessels is required for carbon and low-alloy materials similar to Section I. However, the preheat rules follow the non-mandatory appendix of Section VIII.

Qualification of welding procedures and welders must be in accordance with Section IX. The most important (based upon the discussion it received) additional requirement is the vessel test plates. The vessel test plate requirements are an extension of the requirements in Section I and in Par. UG-84 of Section VIII for impact tested material. The weld and heat affected zone should be comparable to the material in the vessel.

The philosophy of weld examination has been based principally upon radiography during the normal fabrication sequence. It was felt that it would be impractical to require radiography after all fabrication or after postweld treatment and pressure testing has been completed. As a final check, however, subsequent to the pressure test or its equivalent, all weld surfaces of the vessel joints, where physically possible, are to be subjected to magnetic particle examination or to liquid penetrant examination. In those cases where the weld surfaces are not accessible in completed vessels, such surfaces shall be examined immediately prior to the operation which results in this inaccessibility.

## MATERIALS

Safe operation of vessels constructed under this Code has been the primary goal in establishing the requirements for the materials to be used in the construction of such vessels. The special requirements for vessel materials given in this Section are in addition to the normal requirements for the materials employed for vessels in non-nuclear service at equivalent temperatures and pressures. The additional tests and examinations specified are to provide additional assurance of quality and uniformity which is considered necessary because the vessels are generally inaccessible for future internal inspection.

It is intended that these special requirements may be removed from this Section at such time as the material specifications are modified to provide equivalent assurance of quality and uniformity, particularly since the vessels are generally inaccessible for future internal inspection.

Some may consider these requirements to be unduly restrictive and not required or justified to provide adequate assurance of safety, and others may consider the requirements to be inadequate and that the goal should be perfection instead of adequacy. The Committee has considered these aspects of the problem and it is agreed that if, in some cases, the requirements may be unduly restrictive or overly conservative, this is because it is better to err on the safe side; but in no case are these requirements considered inadequate to provide materials of adequate quality for vessels designed and constructed to the rules of this Section of the Code, based on our ability to examine and test materials with the tools that are available today.

These requirements for materials may be supplemented by additional tests and examinations to satisfy specific design requirements for specific applications as deemed necessary or desirable by the designer.

## REFERENCES

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[6] E. E. Baldwin, G. J. Sokol, and L. F. Coffin, Jr., "Cyclic Strain Fatigue Studies on AISI Type 347 Stainless Steel," *ASTM Proc.*, Vol. 57, 1957, p. 567.

[7] L. F. Kooistra and M. M. Lemcoe, "Low Cycle Fatigue Research on Full-Size Pressure Vessels," *Welding Res. Suppl.*, July 1962, p. 297-s.

[8] G. Welter and J. Dubuc, "Fatigue Resistance of Simulated Nozzles in Model Pressure Vessels of T-1 Steel," *Welding J. Res. Suppl.*, August 1962, p. 368-s.

# Appendix D

# ARTICLE I OF SECTION III OF THE ASME BOILER AND PRESSURE VESSEL CODE FOR NUCLEAR VESSELS

## ARTICLE I

## General Requirements

### N-110 SCOPE

The rules in this Section of the Code cover minimum construction requirements for the materials, design, fabrication, inspection and testing, and certification of vessels for use in nuclear power plants.

The Code rules provide minimum safety requirements for new construction and for mechanical and thermal stresses due to cyclic operation. They do not cover deterioration which may occur in service as a result of radiation effects, instability of the material, or the effects of mechanical shock or vibratory loading. These effects shall be taken into account with a view to obtaining the design or the specified life of the vessel.

It is recommended that the increase in the brittle fracture transition temperature due to neutron irradiation be checked periodically by means of surveillance specimens of the same material which receive approximately the same neutron flux as the vessel wall at about the same temperature and neutron energy spectrum. The combined effects of fabrication, stress, and integrated neutron flux should be considered.

### N-120 PARAGRAPH REFERENCES

The main divisions of this Section of the Code are designated as Articles. These are given a number and title, as for example, Article 4, DESIGN. The main division, or subarticles, under these are designated by a three- or four-digit number ending in zero; thus, N-410, Design Criteria. A reference to one of these subarticles in the text of the Section includes all of the applicable rules in that subarticle. Thus, reference to N-410 includes all the rules in N-410 through N-417.7.

The major subdivisions (paragraphs) under subarticles are designated by three-digit numbers which do not end in zero. Further subdivisions (sub-paragraphs) are indicated by a three-digit number followed by a decimal point and a digit other than zero. Further divisions, where necessary, are represented by letters, then by figures, in parentheses. In every case reference to a paragraph includes all the subdivisions under that paragraph.

N-130 CLASSIFICATION OF PRESSURE VESSELS

Vessels covered by this Section of the Code[1] are defined and classified and are related to the Subsections of this Section as follows:

N-131 Class A Vessels - Vessels (or chambers of vessels) not directly open to the atmosphere that have the functions or positions in a nuclear power system defined hereinafter are designated as Class A vessels and shall be designed and constructed in accordance with the rules in Subsection A of this Section.

(a) Any vessel or tubular element that forms part of the reactor coolant system and within which nuclear fuel is present and a nuclear chain reaction may take place.[1] Examples of such vessels are those commonly referred to as reactor vessels.

(b) Any vessel that contains liquid moderator so disposed relative to nuclear fuel embraced within it that a nuclear reaction may take place. Examples of such vessels are those commonly referred to as calendria vessels.

(c) Any vessel or each of a group of vessels which may contain radioactive substance during normal operation and whose function is such that all of the group cannot safely be removed from service when nuclear chain reaction heating is taking place in the system.

(d) Any vessel which may contain radioactive substance during normal operation and for which the specified operation or location is such as to prevent any required periodic examination from being made during plant down time.

(e) Any vessel for which the specified operation is such that the rules in N-415.2 for the analysis for cyclic operation are required to be applied to the vessel or part thereof.

N-132 Class B Vessels - Vessels having a design pressure greater than 5 psi that have the function of position in a nuclear power system defined hereinafter are designated as Class B vessels and shall be designed and constructed in accordance with the rules in Subsection B of this Section.

Metal vessels to operate under positive internal pressure which form structures that enclose vessels, piping, and/or other components of a nuclear system and which are provided for or serve the primary purpose of safely containing or channelling for containment or disposal radioactive or other hazardous effluent which may be released from the vessels or other components so enclosed. Examples of such vessels are those commonly referred to as containment vessels.

N-133 Class C Vessels - Vessels within the scope of Section VIII of the Code that have the functions or positions in a nuclear power system defined hereinafter are designated as Class C vessels and shall be designed and constructed in accordance with the rules in Subsection C of this Section.

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[1]Specifically excluded from consideration in this Section of the Code are tubes or other forms of sheathing used only for cladding nuclear fuel, and also instrument connections 1/2 inch pipe size or less.

(a) Any vessel for which no part is covered by the definitions of N-131 and N-132.

(b) Vessels which are not covered by the definitions of N-131 but which are or may be connected to the reactor coolant or moderator system during operation.

(c) In a multichamber vessel in which one or more chambers are covered by the definitions of N-131, any chambers not so covered may be designed and constructed as Class C vessels.

N-134 Mixed Classes - Any Class B or Class C vessel may be designed, constructed, and stamped in accordance with the rules for Class A vessels.

## N-140 RESPONSIBILITY

The various parties involved in the work of producing vessels under this Section have definite responsibilities in meeting Code requirements. The responsibilities set forth hereinafter relate only to Code compliance and are not to be construed as involving contractual relations or legal liabilities.

N-141 Design Specification - The owner requiring that a vessel or vessels be designed, constructed, tested and certified to be a Code vessel in compliance with these rules shall provide or cause to be provided for each such vessel a specification of functions and design requirements including its classification, related to operating conditions in such detail as will provide a complete basis for design, construction, and inspection in accordance with these rules.

(a) The Design Specification shall be certified as to compliance with the above requirements by a registered Professional Engineer experienced in pressure vessel design.

(b) A copy of the Design Specification shall be filed with the enforcement authority responsible at the point of installation.

N-142 Stress Report - The structural integrity of a Class A or Class B vessel or part thereof including its ability to contain pressure is the responsibility of the manufacturer of the pressure part. A minimum requirement is compliance with the rules of this Section. As part of the design responsibility for pressure containment, the manufacturer or a design agent responsible to him shall make a complete set of stress analysis calculations establishing that the design as shown by the drawings complies with the requirements of this Section for the design conditions that have been specified by the user in the Design Specification. A Stress Report shall be prepared which shall include stress calculations and pressure part design drawings and which shall be certified by a registered Professional Engineer, experienced in pressure vessel design. Copies of this Stress Report shall be filed with the qualified Inspector at the manufacturer's plant and with the enforcement authority at the point of installation of the equipment. The filing of this Report shall not relieve the manufacturer of responsibility for the structural integrity of the vessel. The manufacturer is also responsible for use of materials, fabrication methods, and inspection techniques in accordance with the requirements of this Section. The manufacturer shall certify to compliance with these requirements by the execution of the appropriate Manufacturer's Data Report.

N-143 Inspection Performance - It is the duty of the authorized Code Inspector to make all the inspections specified by the rules of this Section and in addition such other inspections as in his judgment are necessary to verify that the equipment is fabricated in accordance with the requirements of this Section of the Code, including all specific design details necessary for compliance with the requirements of this Section as called for by the design specifications and the design drawings. The Inspector shall so certify on the Manufacturer's Data Report. The Inspector shall not be held responsible for the completeness or correctness of the design calculations as set forth in the Stress Report. The enforcement authority may require the manufacturer to clarify any details of such report if it believes there is reason to do so.

## N-150 BOUNDARY BETWEEN VESSELS AND PIPING

The jurisdiction of this Section of the Code is intended to include only the vessel and its appurtenances and to terminate at the following points where connections are provided for attachment to piping which is external to the vessel:

(a) The first circumferential joint exclusive of the connecting weld in welded connections.

(b) The face of the first flange in bolted flange connections.

(c) The first threaded joint in screwed connections.

Where the external piping differs from Class A vessels in nominal thickness or in coefficient of thermal expansion, the joint shall be analysed for compliance with the stress requirements of this Section of the Code.

Appendix E

# SAFETY STANDARD FOR DESIGN, FABRICATION AND MAINTENANCE OF STEEL CONTAINMENT STRUCTURES FOR STATIONARY NUCLEAR POWER REACTORS*

## INTRODUCTION

Most power reactors are provided with a reactor vessel which encloses the reactor core. This reactor vessel is surrounded by radiation shields for biological protection of personnel.

In addition to this biological protection of personnel, it may be necessary to locate certain reactor types and parts or all of their associated plants, in containment structures so that even under conditions of accidental violation of primary plant integrity, the radioactive material cannot be dispersed in a manner which will harm the public.

The design of such reactor containment must be adapted to each reactor type, as well as the plant location. Present American practice has almost exclusively utilized a steel containment shell for power reactors, and this standard is therefore limited to welded steel shells.

## SECTION 1 - SCOPE

101 - Where it is determined by considerations beyond the scope of this standard that a welded steel pressure containment shell is needed to enclose the reactor installation, this containment structure shall be designed, constructed, tested and maintained according to the requirements of this standard.

102 - This standard assumes that an appropriate design pressure, temperature, volume, configuration and allowable leakage rate have been established after due consideration of all factors involved. (See Section 4)

103 - Large containment structures have, in addition to normal pressure vessel requirements, special requirements imposed by service, structural conditions, and loadings which must be given consideration by the designer. Minimum requirements and their application are set forth in this standard.

## SECTION 2 - APPLICABLE CODES AND STANDARDS

201 - Steel containment shells having a design pressure in excess of 5 psig shall be designed in accordance with the requirements of the

*This standard was approved by the American Standards Association on April 21, 1965. It was prepared by Subcommittee N6.2, Containment, of ASA Sectional Committee N6, Reactor Safety Standards.

ASME Boiler and Pressure Vessel Code, Subsection B of Section III Nuclear Vessels, 1963.* Additional requirements of this standard shall be met.

202 - Steel containment shells having design pressures up to and including 5 psig shall also be designed to meet the requirements of the Nuclear Vessel Code except as otherwise provided in this standard. (See Section 11)

203 - State and local regulations may be more restrictive than this standard and the codes and standards named in Paragraphs 204, 205, 206, and 207 must be considered in the design where they are applicable.

204 - American Institute of Steel Construction, Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, 1961.

205 - American Standard Building Code Requirements for Minimum Design Loads in Building and Other Structures, ASA A58.1-1955.

206 - National Fire Protection Association, Standard, NFPA No 802, Nuclear Reactor, 1960.

207 - American Standard Code for Pressure Piping, ASAB 31.1-1955 with 1963 Addendum and the case rulings in effect as of May, 1963.

## SECTION 3 - MISSILE PROTECTION

Missile protection of the containment shell shall be provided where there are reasonable possibilities that the rupture or failure of any associated equipment, internal or adjacent to the shell, may create missiles which would damage the containment shell. Such protection may be combined with the biological shield.

## SECTION 4 - CONTAINMENT PRESSURE AND LOADING ON CONTAINMENT STRUCTURE

The pressures and temperatures which might be developed within a containment enclosure after an accident will originate from different sources. The rules for the determination of appropriate design pressures and temperatures which have included consideration of equilibrium and dynamic conditions, may appear in other Reactor Safety Standards.

401 - In addition to the stresses created by the internal design pressure, the containment structure shall be designed for any additional loading where applicable such as: loads induced by differential movements caused by variations in temperature between shell and adjacent materials, thermal gradients, weight of the structure, pipe loadings and other concentrated loads, dead loads such as insulation, ladders, platforms, etc., and specified live loads such as maximum loading for floors, air locks, cranes, etc.

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*Hereinafter Section III, Nuclear Vessels, 1963 is referred to as the Nuclear Vessel Code.

402 - The external loads on the containment structure shall include the following, when applicable: the maximum pressure differential between interior and exterior, specified wind or earthquake loads, snow loads, weight of insulation and weight of structure, pressure from external ground water and earth, and concentrated loads. Values for wind loads and snow loads shall be taken in accordance with the requirements of Paragraph 205.

403 - Where in any foreseeable conditions, such as rapid changes in atmosphere or changes in the internal temperature within the shell, the external pressure may exceed the internal pressure by an amount not covered in the design. Vacuum relief devices, of a type such that there is no risk of breach of containment under internal pressure conditions, should be employed.

## SECTION 5 - MATERIAL FOR STEEL CONTAINMENT SHELLS

501 - Since the containment shell is usually the outer housing, the metal temperature is largely dependent on the ambient temperature, and only those materials which remain ductile at the lowest expected metal temperature and all expected operating conditions, and conform to the Nuclear Vessel Code, shall be used for design pressures in excess of 5 psig. For design pressures of 5 psig and under, see Section 11 for exceptions allowing the use of other suitable materials.

## SECTION 6 - ALLOWABLE STRESS VALUES

601 - The primary and secondary stresses resulting from the established combination of loadings as outlined in Section 4 shall be limited to the allowable stress values given by the Nuclear Vessel Code.

602 - Structural members which do not obtain their load from the containment shell pressure or external pressure shall be designed either according to the Nuclear Vessel Code or according to the standard specified in Paragraph 204.

603 - Maximum compressive membrane stresses due to external pressure shall be limited as provided by Paragraphs I-1120, I-1130 and I-1140 of the Nuclear Vessel Code.

604 - Maximum compressive membrane stresses due to loadings in Paragraph 402 (except the external pressure loads for heads and spherical shells covered in Paragraph 704) shall be limited as provided by Paragraph I-1150 of the Nuclear Vessel Code.

## SECTION 7 - SHELL AND HEAD DESIGN

701 - The minimum thickness for a shell shall be determined in accordance with the rules prescribed in the Nuclear Vessel Code, for either

internal or external pressure whichever results in the greater thickness. In those areas where strength requirements are increased by a combination of loadings listed in Section 4, above those required to resist primary membrane stresses due to internal or external pressures, the plate thicknesses shall be increased so as to limit the combined stresses to the values given in Section 6; or as an alternative, where conditions warrant, suitable reinforcement and/or stiffening members may be employed in lieu of increasing the plate thickness.

702 - Individual sections (cylindrical, spherical, conical, toroidal, etc.) shall be analyzed separately for the membrane stress. The junction of two such sections may be treated as a discontinuity using the method of beams on elastic foundations to determine the maximum membrane stresses.

703 - Plate thicknesses for heads shall be determined under the same conditions as specified in Paragraph 701. It should be recognized that the thickness of all dished (except hemispherical) and conical heads subjected to internal pressure may be governed by high circumferential compressive stress at the knuckle or sharp intersection.

704 - In addition to the provisions in Paragraphs 701 and 703, the design of a head or spherical shell shall be checked as follows: The membrane stresses due to the worst combination of loadings as listed in Paragraphs 401 and 402 shall be computed. When both principal stresses are compressive, their numerical difference shall be divided by the allowable compressive stress given by Paragraph I-1150 of the Nuclear Vessel Code assuming a cylinder having a radius equal to the spherical radius, and the smaller of the biaxial compressive stresses shall be divided by an allowable stress equal to one-half of the value of B obtained from Paragraph I-1150 of the Nuclear Vessel Code. The sum of these 2 fractions shall be less than one.

705 - Where internal structures bearing directly against the containment shell provide adequate support, their use to resist elastic buckling of the shell is permissible.

706 - For structures less than fifty feet in diameter, the minimum thickness of shell and head plates shall be 3/16 inch. For structures with diameters from fifty feet to one hundred twenty feet inclusive, the minimum thickness shall be 1/4 inch thick and for vessels with diameters more than one hundred twenty feet but less than two hundred feet, the minimum thickness shall be 5/16 inch. For structures with diameters two hundred feet and over, the minimum thickness shall be 3/8 inch.

707 - Where heavy loads are applied to relatively thin shells, adequate provision should be made to distribute these loads. Consideration should be given to the use of reinforcing pads under structural attachments.

## SECTION 8 - DESIGN OF OPENINGS AND PENETRATIONS

801 - All openings shall be designed in accordance with the requirements of the Nuclear Vessel Code. Where reinforcement is required, the

reinforcement area shall be calculated using the shell thickness required for dead and live loads as well as those for internal and external pressure. Openings shall preferably be circular, but may be elliptical, obround, or rectangular, if required.

802 - The maximum required area of reinforcement shall be applied uniformly around the edge of the opening for circular openings, or for elliptical or obround openings where the ratio of the larger to the smaller dimension of the opening is not greater than 1.5.

803 - In addition to meeting the requirements outlined above, the following limitations shall also apply for large openings with a dimension in excess of 40 inches.

In spherical or ellipsoidal surfaces:

Openings shall preferably be circular. If the openings are not circular, they shall be elliptical or obround and the ratio of the maximum diameter to minimum shall not exceed 1.50.

In cylindrical or conical shells:

Reinforcement for rectangular openings, or elliptical or obround openings having a ratio of the larger to the smaller dimension greater than 1.5, shall be analyzed as a rigid frame. The resulting bending moments shall be combined with the direct stresses due to the pressure loads to determine the reinforcement requirements. Because these forces do not lie in a common plane, it is sometimes necessary to add web stiffeners transverse to the shell around the edge of the opening.

The minimum corner radius of rectangular openings shall be one-third of the maximum dimension or 20 inches, whichever is the smaller.

804 - Access openings for personnel and equipment shall conform to the following:

(a) Openings for use during potentially hazardous operating or non-operating periods, when the full design pressure may occur, shall be in the form of air locks with two doors in series. These doors shall be provided with interlocks to prevent opening of inner and outer doors at the same time. Both doors shall be designed for full design pressure.

(b) Openings which will be used only during potentially hazardous periods when the full design pressure cannot be attained, but when a lesser pressure may be attained, shall also be constructed as air locks with two doors in series, but the doors may be of different types. Inner doors shall be designed for the full design pressure. The outer door may be designed for the lesser pressure. These doors are to be provided with interlocks to prevent unintentional opening of both doors at the same time or to prevent the opening of the inner door when conditions exist that could result in an accident generating a pressure in excess of that for which the outer door is designed.

(c) Openings which will be used only at times when hazardous conditions cannot occur may be constructed as a single door or access panel. Such panel or door shall be designed for the full design pressure.

805 - Penetrations through shell walls for electrical connections shall be reinforced and located in accordance with the principles governing other penetrations. Special consideration shall be given to the design of seals which will permit the passage of electrical power, control and instrumentation cables through the shell wall while remaining leak-tight for the pressure and temperature condition. Consideration shall be given to the possibility of leakage between the cable sheath and the containment shell, between the conductor and the cable insulation, and along the interstices of stranded conductors.

806 - All penetrations for piping and ducts passing through or connecting to the shell shall be designed in accordance with the requirements of this standard. The jurisdiction of standard shall cover piping or ducts between the penetrations in the shell and the first closure outside the shell which can be used to isolate the containment shell in case of an accident. This piping and ductwork shall meet the requirements of the American Standard Code for Pressure Piping, ASA B31.1955 with 1963 Addendum and the case rulings in effect as of May 1963. The design of piping and ducts passing through these penetrations shall include consideration of: material which differs from the containment shell material, differential movements, operating temperature, shielding, type of welds and possible methods of inspection.

807 - All piping and ducts which penetrate the containment shell shall be designed to prevent hazardous release of radioactive material to the atmosphere in the event of a failure of this piping and ducts or an incident inside of the containment shell. The piping and ducts which are open to the atmosphere and penetrate the containment shell must treated as ventilation ducts, covered in Section 18.

## SECTION 9 - SPACING OF OPENINGS

901 - The distance between the centers of adjacent individually reinforced openings shall be at least equal to the sum of the two finished opening dimensions which lie in the same transverse plane. Where so-called insert fittings are used such that they require butt-welding into the shell in the field, this dimension shall be increased an additional 12 inches.

902 - Common reinforcing plates may be used if stress relieved as a unit. The spacing between the centers of adjacent openings in a common insert shall be not less than 3/4 the sum of the two finished opening dimensions.

903 - When one of the adjacent finished openings is greater than 40 inches in any dimension, the weld around its insert reinforcement shall be not less than 36 inches or the diameter of the smaller opening

whichever is smaller from the weld around the insert reinforcement of the adjacent penetration.

904 - Field welds around inserts shall preferably be located to clear main vessel welds by 12 inches or more; where such clearance is not possible inserts must be located so that the weld shall cross main welds with an intersection of the insert weld and any main weld at an angle of not less than 30 degrees.

905 - The distance between adjacent fillet welds attaching doubler plates or between a fillet weld and a main seam or a discontinuity (head to shell junction or stiffener) shall be eight times the nominal fillet weld size (where two fillet welds are involved, the larger shall apply) or 6 inches whichever is greater.

## SECTION 10 - WELDING

1001 - The welding and welding inspection shall meet the requirements of both the Nuclear Vessel Code and Section IX of the ASME Boiler and Pressure Vessel Code, Welding Qualifications, 1962 with case rulings in effect as of May 1963.

1002 - Because these shells are usually field erected, the welding procedure qualifications include a variety of the welding positions defined in the above sections of the Code, as well as various types of joints and may include both automatic and manual welding.

1003 - When making field butt welds between insert weldments and shell, special procedure may be required to prevent distortion. These may include preheating and temporary shell stiffening.

## SECTION 11 - REQUIREMENTS FOR STEEL CONTAINMENT SHELLS DESIGNED FOR PRESSURES OF 5 PSIG OR LESS

1101 - Steel containment shells having design pressures up to and including 5 psig shall meet the requirements of other paragraphs in this standard, except as modified in this section.

1102 - Radiographing of longitudinal and circumferential welded joints and reinforcements may be reduced to spot examination of welded joints as covered in Paragraph UW-52* of Section VIII of the ASME Boiler and Pressure Vessel Code, Unfired Pressure Vessels, 1962 (referred to as the UPV Code), using the appropriate joint efficiency. The acceptability of welds examined by spot radiography shall be judged by the requirements of Paragraph UW-52 except that the requirements of Paragraph UW-51-(m)(2) and (m)(3) covering limits on slag inclusions shall replace the requirements of Paragraph UW-52(c)(2) covering slag inclusions and cavities.

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*References to paragraphs beginning with U are to the UPV Code.

1103 - Plate material according to ASME Specifications SA-201 ordered to fine grain practice, SA-442 ordered to fine grain practice, or ASTM Specification A 131-59 Grade C, may be used for design metal temperatures and thicknesses as follows:

| Plate Thickness | Design Metal Temperature |
|---|---|
| Not over 1 3/8" | -5°F |
| Not over 1/2" | -20°F |

Note the design metal temperature may be assumed to be 15°F higher than the lowest recorded one day mean temperature as shown in Appendix A.

1104 - The allowable stress value for ASTM Specification A 131-59 Grade C material shall be 12,650 psi for temperatures between 650°F and the minimum specified in Paragraph 1103.

1105 - Forgings, pipes and tubes according to ASME specifications SA-181, or SA-106 which form part of the containment structure may be used for design metal temperatures and thicknesses as specified in Paragraph 1103.

1106 - Minimum plate thickness shall follow the requirements of Paragraph 706.

1107 - Openings shall meet the requirements of Sections 8 and 9 of this standard and the following, whichever results in the maximum reinforcement. Openings shall be reinforced 50 percent based on the nominal plate thickness required for stability as required in Paragraph 603 or 706. Openings smaller than 12 inches need not meet this 50 percent requirement.

1108 - For materials permitted by Paragraphs 1103 and 1105, impact tests on vessel test plates shall be made for the weld metal only.

1109 - All welded frames for openings shall preferably be stress relieved before welding into the shell. This is mandatory when any plate in the frame, used for reinforcement, exceeds 5/8 inch.

## SECTION 12 - STEEL LININGS FOR PARTS OF CONTAINMENT STRUCTURES MADE OF OTHER MATERIALS

1201 - For parts of containment structures where the steel plate serves primarily as a membrane liner to reduce leakage and where stresses due to pressure or temperature are not a consideration, single lap welds up to 3/8 inch thickness may be used.

1202 - Weldable quality structural plate material is satisfactory.

1203 - Radiographing of seams in liner plates will not be required.

## SECTION 13 - CONTAINMENT INSULATION

The benefit of thermal insulation will vary for each installation and should be determined for each case. In a cold climate external insulation will prevent internal condensation on the walls, reduce the heating or air conditioning load and reduce thermal stresses due to variable weather exposure, and also reduce the possibility of brittle fracture.

Precaution shall be taken to prevent corrosion of the shell.

## SECTION 14 - FOUNDATION AND SUPPORT REQUIREMENTS DURING ERECTION AND OPERATION

1401 - Due to the critical nature of a nuclear installation, more than the usual amount of attention must be given to the evaluation of normal structural deflections and slight movements of the structure and the load. Particular attention shall be given to the possibility of earthquakes and differential settlement.

1402 - Water pressure from external ground water or internal flooding should be considered in both the foundation design and the shell design.

1403 - Where the permanent foundation consists of a concrete slab poured under the shell, extreme care must be taken to eliminate, as far as possible, any gap between the slab and the contacting surface of the shell to minimize distortion and movement.

1404 - Where soil backfill is placed directly against the steel shell, suitable corrosion protection shall be provided.

## SECTION 15 - PRESSURE TESTING FOR STRENGTH

1501 - Containment shells shall be pressure tested in accordance with the Nuclear Vessel Code.

1502 - No permanent distortion shall occur as a result of the test.

1503 - All penetrations and seals on the containment shell shall withstand a pressure test.

For access openings having two closures in series which are interlocked so that neither can be opened unless the other if fully closed and secured, in accordance with Section 8, each closure shall be pressure tested separately to demonstrate that each can withstand its appropriate test pressure.

1504 - The pressure test shall be carried out after completion of all construction affecting the containment shell unless pouring of concrete or other construction would make portions of the shell inaccessible for testing and repair.

In cases where subsequent construction may affect shell integrity a final test shall be carried out after completion.

If the work on the containment shell subsequent to the initial pressure test is of such limited scope that it can be reliably shown by engineering analysis that it does not affect the integrity of the overall shell, it is permissible to omit the second pressure test provided that any new welds and seals are separately inspected for soundness and tightness by code-recognized methods.

## SECTION 16 - LEAKAGE TESTING

1601 - The following kinds of leakage tests shall be performed:
(a) Local leakage tests.
(b) Integrated leak-rate tests.

The function of the local tests is to locate any leaks of detectable size at any of the places where leakage could reasonably occur.

The function of the integrated leak-rate test is to make sure that the overall leak-rate does not exceed the allowable value.

1602 - Local Leakage Tests: All welds, seals and other types of joints shall be tested for leak-tightness by the soap bubble method or other method with at least equivalent sensitivity. Use of a halogen compound vapor inside the containment shell with halogen detectors used to examine each weld, seal, or other joints is an example of such an alternative method. A combination of such methods may also be used.

If the local leakage test is carried out as an internal pressure test, then a pressure of at least 5 psig shall be used if the design pressure of the containment shell is above 10 psig; and at least 1/2 of the design pressure if the design pressure is 10 psig or less.

1603 - Initial Integrated Leak-Rate Test: The initial integrated leak-rate test consists of pneumatic pressurization of the containment shell to between 85 percent and 100 percent of its design pressure and observation of the changes of that pressure with time, to determine how much air, if any, leaks out of the vessel in a given time.

The pressure during this test shall be measured with sufficient accuracy and over a long enough period of time to insure that the amount of leakage that has taken place is less than the allowable leakage rate. In determining whether the allowable leakage rate has not been exceeded, allowances must be made for the effects of temperature and humidity variations on the pressure readings.

1604 - Either of the following methods may be used for leakage rate testing.
(a) Pressure, temperature and humidity measurements at a number of points inside the containment shell shall be taken over a long enough period to obtain adequate reliability.
(b) Determination of any pressure changes in the containment shell may be taken with respect to a reference system consisting of one or several containers distributed within the containment shell. At the beginning and end of the selected test period for determination of leakage, the average temperature of the air in the reference system and of the air in the containment

shell shall be approximately equal. The period from midnight to dawn usually offers the most stable temperature conditions. If such a reference system is used, the degree of leak-tightness of the reference system itself must be established.

1605 - Penetrations and seals on the containment shell shall be subjected to the local leakage tests and shall normally be exposed to the integrated leak-rate test. When there are two operable seals in series on a penetration, such as access and ventilation openings, then one seal shall be open during the leak-rate test and the other shall be closed. The integrated leak-rate test need not be repeated with the open-and-closed situation reversed if both the seals in the series have successfully withstood the pressure test and revealed no imperfections in the local leakage tests. Minor penetrations added as a result of new work shall be subjected to local leakage test but need not be subjected to integrated leak-rate tests.

## SECTION 17 - PERIODIC INSPECTION AND TESTING

1701 - The purpose of periodic inspection and testing is to insure that the allowable leakage rate is not exceeded during the life of the plant.

The pressure used for periodic leakage testing shall be as close as possible to that used for the Initial Leak-Rate Test but it is often limited by the fact that it is difficult to subject the final installation to full pressure tests because some instrumentation and equipment might suffer.

1702 - Most leakage will normally occur through large access openings, air locks, valves, etc., which are subject to use during normal operations, and it is recommended that these be tested at least once a year. It is recommended that other removable covers be tested after each period of use in which they are removed.

Integrated leakage rate retest shall be required only after significant repairs on the containment structure have been made, or if excessive corrosion or other deteriorative processes are evident from the annual inspection. Such a test shall be performed at least once within the first five years of reactor operation.

1703 - The continuous structural integrity and leak-tightness might be jeopardized by the following conditions which should be subject to an annual inspection:

(a) Unequal settlement of the foundations.
(b) Corrosion.
(c) Deterioration with consequent leakage at a connection, door, or removable cover.
(d) New work on the containment shell.
(e) Mechanical impact damage.
(f) Cracking at points of stress concentrations.

1704 - Records shall be kept of all inspections and tests required by this section.

## SECTION 18 - VENTILATION AND INTERLOCKING EQUIPMENT

1801 - Ventilation ducts penetrating the containment shell shall be designed for the same conditions, i.e., pressure, temperature, leakage, etc., as the containment shell.

Where ventilation ducts penetrate the shell and for cases where the ventilation system operates during periods when potentially hazardous conditions exist, both inlet and exhaust ducts shall contain automatic quick-closing isolation valves. For cases where such a ventilation system will never be operated during such potentially hazardous periods, isolation valves may be either automatic or manual, but shall be interlocked to prevent reactor operation or other potentially hazardous activities unless all such valves are fully closed.

1802 - The operating system for automatic quick-closing isolation valves shall be so designed that the failure of any single component, which has a reasonable probability of failure, cannot affect the integrity of the containment vessel. A single valve system will be satisfactory, if an alternate emergency arrangement is provided to ensure that the system meets this requirement.

1803 - Isolation valves which are normally open during potentially hazardous periods shall be controlled to close automatically in case any of various indications of abnormal behavior within the containment exceeds a permissible level.

Interlocks shall prevent the accidental reopening of these valves by operating personnel until conditions make it safe to do so.

The leakage through a closed ventilation isolation valve with containment at design pressure shall be included in the integrated rate for the entire containment shell.

## SECTION 19 - FIRE PROTECTION AND GENERAL SAFETY

1901 - For consideration of fire protection, reference is to be made to National Fire Protection Association Standard, NFPA N-802, Nuclear Reactors, 1960.

1902 - Some means of communication shall be provided between air lock and control room (or appropriate outside area) to prevent the possible trapping of personnel in the air lock should doors fail to function properly.

# Appendix F

## STANDARD ACCOUNTS

Standard accounts for all significant cost items in a nuclear plant are presented in the AEC document TID-7025, Guide to Nuclear Power Cost Evaluation. Account 219, Reactor Containment Structure, from this document is reproduced below:

219 Reactor Containment Structure

This account is applicable for nuclear power plants utilizing pressure containment type structures. Refer to Account 212G, Reactor Building, for plants that do not come under this category.

(Expand subdivisions as necessary).

.1 Excavation and backfill
  Excavation
    Earth
    Rock
  Backfill
  Sheeting and shoring
  Dewatering
.2 Bearing piles and caissons
.3 Substructure concrete
  Forms
  Reinforcing
  Concrete
  Waterproofing
  Patch and finish
  Miscellaneous anchor bolts, sleeves, etc., embedded in concrete
.4 Superstructure
  .41 Structural steel
  .42 Containment shell
    Shell
    Testing
    Air locks
  .43 Exterior insulation and painting
  .44 Floors, barriers and internal concrete, excluding biological shielding and equipment foundations
  .45 Interior finish
    Miscellaneous iron and metal
      Stairs and platforms
      Grating and checkered plate
      Embedded iron
      Miscellaneous metal
.5 (Reserved)

.6 Building services

Plumbing and drainage system

- Plumbing
- Drainage
- Sump pump

Heating system (closed)

Ventilating system

- Closed cooling system (usually combined with closed heating system)
  - Fans
  - Ductwork
  - Service water
  - Steam piping, including valves
  - Pipe covering
- Purging system
  - Fans
  - Valves
  - Air filters and coils
  - Ductwork
- Air conditioning system (not usually provided)
- Elevators (optional, not usually provided)
- Lighting and service conduit and wiring, including control panels
- Fire protection (water lines, hose, sprinklers, etc.)

Appendix G

# BEHAVIOR OF IODINE IN REACTOR CONTAINMENT SYSTEMS

(Editor's note: This appendix was adopted from a report[1] of the same title recently released by the Nuclear Safety Information Center; the report was prepared by G. W. Keilholtz and C. J. Barton of the ORNL Reactor Chemistry Division. The excerpted material is preceded by an introduction written by R. R. Newton of the USAEC Division of Reactor Development and Technology.)

## 1. INTRODUCTION

One of the major criteria[2] used in the siting of water-cooled power reactors today is the potential hazard of the fission-product release that could result from a loss-of-coolant accident. The lack of definitive information on this subject forces us to use conservative assumptions that impose economic penalties on power reactors by siting them large distances from the population centers they are to serve. Two approaches are being taken to reduce this penalty. The first is to better characterize the chemical and physical forms of accident-released fission products, together with their behavior (transport, plateout, etc.), in order to minimize what may be unduly conservative assumptions. The second is to develop and test both "passive" and "active" engineered safeguards to limit the consequences of a serious accident.

Although all types of released radioactivity are of some concern in safety analyses, it is generally considered that the chief hazard to the public from a catastrophic accident (in a water-cooled and -moderated reactor) would be due to fission-product iodine; hence the behavior of accident-released fission-product iodine controls the present site criteria. The potentially high hazard associated with fission-product iodine is due to (1) the high fission yields of the radioiodines, (2) the high release fractions of iodine found on the destruction of fuel, (3) the high degree of mobility of molecular iodine and many iodine compounds of interest, and (4) the high radiotoxicity of radioiodine, especially $^{131}I$.

Experiments have shown that the behavior of accident-released fission-product iodine is exceedingly complex.[3-10] Even for a given reactor, a loss-of-coolant accident would result in a variety of physical and chemical forms of fission-product iodine that would depend on the exact nature of the accident, including the complex and constantly changing combinations of environmental conditions that might exist at a given location at a given time. For example, a pipe rupture that allowed fairly rapid air inleakage into the core region would tend to cause larger fractions of the released fission-product iodine to appear as molecular iodine.[11] Alternatively, a top-pile rupture several feet from the pressure vessel wall could promote the production of HI at the expense of other forms,[12] since the conditions of high core temperature and residual coolant remaining after the top rupture would lead to considerable hydrogen production via metal-water reactions. The relative amounts of $I_2$ and HI could be important, since HI is more soluble in water than $I_2$ and thus more susceptible to being carried by fog droplets or to being scavenged by engineered water-spray safeguards than is $I_2$. As an indication of the complexity of the problem it should be noted that these conditions (oxidizing and reducing environments) are not mutually exclusive in the core region; these and other conditions can occur simultaneously. Also, sizable fractions of accident-released iodine would tend to physically adsorb (both reversibly and irreversibly)[3,13] on airborne particulates (aerosols) present either as normal condensation nuclei or as meltdown debris. Likewise, chemical reactions with these particulates could be reversible and irreversible.

It should be noted that the initial form to which iodine would tend immediately after its release would not necessarily be its final form. Further, the initial form would not necessarily be the form that would have the most significance relative to the transport characteristics, the effectiveness of engineered safeguards, or the overall "hazard" of the accident. It is necessary to know the changes through which iodine forms may proceed as a function of time and conditions in order to remove possible undue pessimism regarding accident consequences and to design and test adequate engineered safeguards. It must be considered that the probable initial form of release from the core (very high temperature zone) is atomic iodine. This iodine would become exposed to different temperatures in different environments: mixtures of steam, hydrogen, air, and other gaseous compounds and aerosols. The exact nature of the environment the iodine could see would depend on its point of release, the exact past course of the accident, and the phenomena that had previously occurred and were simultaneously occurring in other parts of the core. For instance, the core temperature gradient might cause iodine to be released from melting fuel at a time when other parts of the core were only hot enough to contribute aerosols to the environment by melting the cladding. It should also be noted that the iodine form could depend upon the release temperature, which, in turn, would be dependent on both the position of the fuel pin and the time since the accident was initiated. For instance, the core heatup environment would not be homogeneous; a given location could go through the following: heatup up to about 1500°C to melt the cladding, iodine release primarily by diffusion up to the fuel melting temperature of 2750°C, almost quantitative release of remaining iodine from the molten fuel, and release by oxidation once the local temperature fell back below 1500°C. Release via oxidation could be particularly important for fuel pins that reached cladding-melting but not fuel-melting temperatures. Different phases of such sequences could occur at different core locations at any given time. In addition, melting and oxidation of cladding or fuel, followed by run-off, puddling, or slumping, might shift core materials to different temperature zones during the course of an accident.

At the present time, core meltdown analyses and hence fission-product release models are based largely on speculation, since there is not even much qualitative information available, and thus may result in unduly conservative siting requirements. It is not known, for example, to what extent molten cladding (or cladding oxide due to metal-steam reactions) resulting from a loss of coolant in a water-cooled power reactor would significantly "wet" the $UO_2$ fuel. There is some evidence[14] of cladding oxide-$UO_2$ eutectic formation that lowers the fuel melting-point over 800°C. Such melting-point lowering could significantly increase the rate of iodine evolution. There is other evidence[15] which indicates that at some heating rates the stainless steel oxide that forms via reaction with steam is much less dense than the original stainless steel. Thus "swelling" of stainless steel core components upon oxidation might tend to keep the core in place for a longer time after blowdown than is currently assumed; that is, core slumping could occur at a much later time. The formation of a "fluffy" oxide might act as a fission-product trapping medium. Further, it is currently assumed that the cladding at the peak flux area of a given fuel rod would be the first to melt (either as the metal or the oxide). However, it is not clear that this would allow fuel pellets above this area to slide out of the cladding (either metal or metal oxide or both) at this point, or that they would remain in the sheath because of some sort of chemical interaction between fuel pellets or between the fuel and the cladding (or cladding oxide); diffusional release of fission products would be higher from unclad fuel than from fuel that remained in its sheath. If very hot $UO_2$ fuel were to fall out of its sheath and thus be exposed to steam, there is some evidence[16] that it would be partially oxidized with a further evolution of hydrogen (and presumably fission products). Yet another factor that is still largely unknown is the effect of past operating temperature of the fuel; the creation of void spaces for fission-product-gas pressure buildup via diffusion could cause cladding to burst following a blowdown accident. Again we must speculate on what would happen to the fuel pellets and gaseous fission products. This type of behavior could also result in extensive fragmentation that would thus increase the extent and rate of metal-water reactions. All these events would probably tend to produce extremely complex mixtures of iodine forms in the core region.

As the iodine plated out or was transported out of the primary system, interchanges of iodine could occur as the environment changed to the predominently steam-air lower temperature (to 300°C) volume of the containment. Consequently, changes of form with time may be expected. Indeed, even iodine that plates out can experience chemical changes with surface materials and impurities (as well as normal atmospheric impurities) and reevolve from the surface for further transport.[17] Two groups of organic compounds[3,8,17] have, in fact, been observed; one group is composed primarily of alkyl iodides, while the other is as yet unidentified. (These organic iodides have particular significance because of their ability to penetrate charcoal filters under certain conditions.[17,18]) Much of the anomalous behavior of accident-released iodine is due to its relatively high chemical reactivity, coupled with the relatively low concentrations of iodine involved. Average total iodine concentrations between virtually zero and 500 mg/m$^3$ of containment volume would be possible following a loss-of-coolant accident in a large power reactor, depending on such assumptions as the degree of burnup and the extent of release from the primary system. The fraction of iodine converted to organic forms (such as $CH_3I$) seems to decrease with iodine concentration, at least in clean laboratory systems.[19]

It is important to learn how the relative amounts of these various iodine species vary as a function of (1) position in both the containment shell and primary system and (2) time-after-release for various release conditions for the following reasons:

1. The physical and chemical form assumed by the iodine will determine its transport and plateout characteristics in the primary system and containment shell and hence determine to what extent it remains available for leakage to the environs. One of the important questions is the reversibility or irreversibility of absorption of the various forms on surfaces of interest. Unless irreversibly adsorbed, material may be available for subsequent release as accident conditions change.

2. The physical and chemical form assumed by the iodine must be known as a function of time in order to develop effective engineered safeguards, such as air cleanup systems. There is some evidence,[17,18] for example, that current air-cleaning systems do not have high removal efficiencies for methyl iodide, particularly at intermediate temperatures and in steam atmospheres (efficiency apparently changes as a function of temperature).

3. The physical and chemical form of the iodine released to the environs determines its deposition velocity[20] and hence its concentration profile.

4. The physical and chemical form of radioiodine determines its biological hazard. For example, the MPC (ref. 21) of $^{131}I$ in water in a restricted area is only $9 \times 10^{-9}$ μc/ml for soluble forms but $9 \times 10^{-7}$ μc/ml for insoluble forms.

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# 2. PHYSICAL AND CHEMICAL PROPERTIES OF IODINE AND IODINE COMPOUNDS*

## 2.1 PHYSICAL PROPERTIES OF ELEMENTAL IODINE

Iodine is a black solid with a slight metallic luster. Gaseous iodine dissociates more readily than the other halogens. Free energy values for the dissociation are given in Table 2.1. Under 1 atm of pressure, dissociation is said (*7*) to become perceptible at 600°C, 5.2% is dissociated at 800°C, 19.7% at 1000°C, and 74.8% at 1400°C.

Table 2.1. Calculated Free Energy Values for the Reaction $\frac{1}{2}I_2 \rightarrow I$ (*7*)

| Temperature (°K) | $\Delta F$ (kcal/mole) |
|---|---|
| 298 | +14.44 |
| 500 | +11.98 |
| 1000 | +5.74 |
| 1500 | −0.62 |
| 2000 | −7.10 |

Table 2.2. Isotopes of Iodine Produced in Reactors (*1*)

| Isotope | Half-Life | Fission Yield[a] (%) | Activity per Unit Thermal Power (kilocuries/Mw) At Shutdown | 1 Day After Shutdown |
|---|---|---|---|---|
| $^{131}I$ | 8d | 3.1 | 25 | 23 |
| $^{132}I$ | 2.3h | 4.7[b] | 38[c] | 0 |
| $^{133}I$ | 21h | 6.9 | 54 | 25 |
| $^{134}I$ | 52m | 7.8 | 63 | 0 |
| $^{135}I$ | 6.7h | 6.1 | 55 | 4.4 |
| $^{136}I$ | 86s | 3.1 | 53 | 0 |

[a]Fission yield values from Katcoff (*9*) for thermal neutron fission of $^{235}U$.

[b]Yield of $^{132}Te$.

[c]Amount of $^{132}I$ generated in the reactor by decay of $^{132}Te$ and yield of this isotope.

Iodine melts at 113.6°C and boils at 184.4°. The vapor pressure of the solid is represented (*8*) by the equation

$$\log P_{atm} = -3512.8/T + 2.013 \log T + 13.3740 .$$

Natural iodine ($^{127}I$) has an absorption cross section of 6.3 barns. The principal fission-product isotopes of iodine are listed in Table 2.2. All are beta emitters.

## 2.2 CHEMICAL PROPERTIES OF IODINE

Information in this section is drawn mainly from reference books on nonradioactive iodine or on the halogen family (*7–8, 10*). Certain aspects of iodine chemistry that are more or less unique to the field of fission-product behavior, mainly because of the radioactivity and the small mass amounts of material involved, are treated in the following sections.

Iodine, along with the other members of the halogen family, is placed in Group VIIA of the Mendeleev periodic chart. Its outermost electron shell has seven electrons, but its *N* shell has only 18 electrons out of the maximum of 32. It has the lowest electronegativity among the naturally occurring halogens, and it exhibits valence states of −1, +1, +3, +4, +5, and +7 in various compounds.

### 2.2.1 Molecular Iodine, $I_2$

The chemical activity of iodine is less than that of the other halogens. There is no appreciable reaction with either hydrogen or oxygen at ordinary temperatures. Its reaction with metals is discussed in Sect. 2.2.2. The standard redox potential ($E_0$, volts) for the reaction

$$I_2 + 2e^- \rightarrow 2I^-$$

is 0.5345, as compared with 1.087 for $Br_2$, 1.3583 for $Cl_2$, and 2.85 for $F_2$.

When $I_2$ is dissolved in water (0.029 g per 100 g of $H_2O$ at 20°C) slight hydrolysis occurs according to the equation

$$I_2 + H_2O \rightarrow HIO + H^+ + I^- ;$$

*The references cited in this excerpted material are to be found in the appendix Bibliography.

the hydrolysis constant for this equation has been calculated to be $4.6 \times 10^{-13}$ at 25°C.

Iodine oxidizes thiosulfate ion, $S_2O_3^{2-}$, to $S_4O_6^{2-}$ in a bicarbonate-buffered solution or to $SO_4^{2-}$ in a strongly alkaline solution. These reactions are the basis for proposals to add thiosulfate to dousing solutions.

### 2.2.2 Metal Iodides

Fission-product iodine released in reactor accidents is likely to come into contact with metal surfaces; consequently, the stability of metal iodides is of interest. This subject has been discussed at some length by Rolsten (*8*). The following elements form sufficiently volatile iodides to be amenable to production in a pure form by the iodine process: Y, Ti, Zr, Hf, Th, V, Ni, U, Nb, Ta, Cr, W, Cu, Ag, Fe, B, Ge, Pa, and Si. It will be noted that the three principal components of stainless steel, iron, nickel, and chromium, are included in this list. This is important because stainless steel is a common material of construction in nuclear reactors, especially of primary reactor vessels in pressurized water reactors. The standard heats of formation of anhydrous iron, chromium, and nickel iodides are −30.0, −54.2, and −20.5 kcal/mole, respectively.

Earlier work on reactions of iodine with metal surfaces is summarized by Mellor (*10*). This subject is being re-examined at the Oak Ridge National Laboratory by use of tracer techniques, with special emphasis on interaction of gaseous iodine with stainless steel surfaces. Preliminary results of this continuing investigation have been reported (*11–13*). Iodine vapor densities varied from ~0.001 to 1000 mg/m$^3$, and the temperature of the metal surface (types 302, 304, and 321 stainless steel) varied from 24 to 700°C. No difference in behavior toward iodine vapor was exhibited by the three different types of steel employed. It appears that the reaction probably occurs at points on the steel surface where imperfections exist in the thin oxide coating that normally covers such surfaces. The resulting iodide product deliquesces in the presence of moist air and will eventually lose iodine with the production of oxide. Surface coverages of up to 60 monolayers (a monolayer is approximately 0.3 $\mu$g of $I_2$ per cm$^2$) were observed, but large variation in the saturation coverage was also noted. Results of exploratory experiments (*11–13*) are summarized in Table 2.3. The information in this table indicates that iodine reacts with stainless steel surfaces over the temperature range 25 to 700°C and that two components exist in the adsorbed phase, one more readily removable than the other. Data (*14*) on the kinetics of iodine sorption by stainless steel are summarized in Table 2.4, and adsorption by preoxidized steel is compared with that by as-received stainless in Table 2.5. The latter data show that the oxidized steel adsorbed only 3% as much as the as-received specimens with a 1000-fold increase in $I_2$ concentration. Reactions occurred in an essentially oxygen-free atmosphere in this work, and application of the results to accident situations where oxygen and steam will probably be present is not clear. Reactions of iodine with metal surfaces in the presence of oxygen and steam as well as reactions of iodides on surfaces with oxidizing atmospheres will be of more direct interest for hazards analyses.

### 2.2.3 Hydriodic Acid, HI

Hydrolysis of molecular iodine, mentioned in Sect. 2.2.1 above, results in production of HI and HIO. Properties of hydrogen iodide are considered briefly in this section. It is a strong reducing agent and is extremely soluble in water. Solid HI melts at −50.8°C, and the liquid boils at −35.4°C. A maximum boiling mixture of aqueous HI contains 53% HI at 1 atm and 127°C. Thermal decomposition according to the equation $2HI \rightarrow H_2 + I_2$ is represented (*7*) by the formula

$$\log K_{HI} = -10{,}030/T + 0.5 \log T + 13.001 \ .$$

It is a strong acid but not as strong as HBr or HCl. It reacts with active metals, oxides, hydroxides, and carbonates to form iodides.

### 2.2.4 Oxy Compounds of Iodine

Hypoiodous acid, HIO, is formed by hydrolysis of $I_2$ as indicated in Sect. 2.2.1, and it is also formed when iodine is dissolved in cold dilute alkali solutions. On standing or on heating, the hypoiodite ion decomposes to give the more stable iodate ion.

The amphoteric nature of HIO is represented by the equation: $HIO \rightleftharpoons H^+ + OI^- \rightleftharpoons OH^- + I^+$.

**Table 2.3. Experiments Conducted on Iodine Adsorption (*11–12*)**

| Type of Experiment | $I_2$ Vapor Densities ($mg/m^3$) | Adsorbent Temperature (°C) | Principal Findings |
|---|---|---|---|
| Closed loop; helium carrier gas; in situ gamma-ray detection of $^{131}I$ tracer on type 304 stainless steel | $\sim 10^{-3}$ and $10^2$ | 24–700 | Iodine initially reacted irreversibly with SS surfaces and was not further transported, even during 3 weeks' operation. |
| Metal coupons in evacuated glass apparatus; intermittent removal of coupons and chemical analysis for iodine in leached deposit | $\sim 10^{-3}$ to $10^3$ | 24–500 | Water-soluble iodides formed on SS surface up to approximately 60 monolayers. Introduction of air seemed to cause loss of iodine from the surface deposit. The amount of deposit of iodine on the metals varies in the order Cu > Fe > Ni. |
| Metal coupons in evacuated glass apparatus; in situ gamma-ray detection of $^{131}I$ tracer on surface | $\sim 10^{-3}$ to $10^2$ | 24–260 | Activity sorbed on the glass surface masked activity on the SS at low vapor concentrations. Two components in the adsorbed phase, one readily reversible and one rather irreversible, are observed at higher pressures of $I_2$. Reversible fraction decreased with time. |
| Type 304 stainless steel tube coupled to glass vacuum-source system; in situ gamma-ray detection of $^{131}I$ tracer on SS surface | $\sim 10^{-3}$ to $10^2$ | 300–500 | Sorption of only a few monolayers, reversible to some extent, leaving SS surface passive toward re-adsorption. Eventually passivity was overcome at higher iodine pressures. A second test section was passive toward iodine adsorption from the start of the experiment. Investigation on this sample is in progress. Mass spectrographic analysis of source indicates presence of nonelemental forms of iodine. |
| Temperature gradient imposed along type 321 stainless steel tube coupled to glass vacuum-source system; in situ gamma-ray detection of $^{131}I$ tracer on SS surface; detector on moving platform so as to traverse the temperature gradient region of the tube | $\sim 10^{-3}$ and 1 | 24–380 | Maximum adsorption at 150°C; no sorption in hot zone even on cooling (passivation had apparently taken place). On changing the temperature profile along tube, material moved so as to maintain the maximum in the 150°C region of the tube. |

Table 2.4. Summary of Experimental Information on Kinetics of Iodine Sorption by Stainless Steel

| Surface Temperature ($^{\circ}C$) | Iodine Concentration Above Surface ($mg/m^3$) | Rate Coefficient, $\alpha$ (monolayers$^{-1}$) | Time Range (min) | Number of Monolayers |
|---|---|---|---|---|
| 70 | 0.006 | 40 | 10–100 | 0.06 |
| | | 9 | 100–400 | 0.26 |
| 70 | 0.166 | 2.7 | 100–1000 | 2.1 |
| 180 | 0.126 | 0.7 | 90–400 | 3.0 |
| 180[a] | 0.99 | 9 | 1–100 | 0.09 |
| 500 | 0.00270 | 1.5 | 10–100 | 2.3 |

[a]Surface pretreated to have an oxide layer 200–300 A thick.

Table 2.5. Effect of Stainless Steel Surface Oxidation on Sorption of Iodine Vapor at a Surface Temperature of 150°C

| | As-Received Stainless | Preoxidized Stainless |
|---|---|---|
| Surface oxide characteristics | Amorphous, undeveloped, probably thin | 200–300 A, crystalline |
| Iodine pressure (mm Hg) | $1.4 \times 10^{-5}$ | $1.1 \times 10^{-2}$ |
| Time exposed (hr) | 8.1 | 12 |
| Monolayers sorbed | 4.5 | 0.14 |

Hypoiodous acid is a very weak acid, as shown by its ionization constant, $2 \times 10^{-10}$.

Iodic acid, $HIO_3$, with a heat of formation of 56 kcal/mole, is more stable than either $HClO_3$ (24 kcal) or $HBrO_3$ (12.5 kcal). Iodic acid solutions can be concentrated until crystals precipitate.

### 2.2.5 Methyl Iodide

A number of organic compounds are apparently formed when radioactive iodine is released in the atmosphere of containment vessels and when molecular iodine is generated in laboratory tests (see Sect. 3.6). Methyl iodide forms the major portion of the group of aliphatic iodides that have been identified, and it is the only one of the group that is considered here.

Methyl iodide (iodomethane) boils at 42.5°C (760 mm) and freezes at –66°. It is available commercially tagged with $^{131}I$ and possibly can be obtained with other radioiodine isotopes, so it is unlikely that many investigators will make their own tagged methyl iodide. However, several methods for this synthesis are discussed in the literature. Since $^{131}I$ is most readily available in the form of $Na^{131}I$, possibly the most convenient method is exchange of this compound with $CH_3{}^{127}I$ in ethanol (*15*).

Methyl iodide is slightly soluble in water (1.4 g per 100 g of water at 20°C). The standard free energy of formation of the compound from its elements ($\Delta F_{f0}$) at 25°C is listed (*16*) as –5.3 kcal/mole.

Gover and Willard (*17*) have studied the effect of short-wavelength light (1849 A) on the reaction of hydrocarbons with iodine, but similar studies of the effect of radiation on such reactions have not come to the attention of the authors.

# 3. GENERATION AND CHARACTERIZATION OF FISSION-PRODUCT IODINE AND OF PARTICLES

The characterization of fission-product iodine from a fuel melt is difficult because of the wide ranges of temperatures, fuels, contaminants, and environments which may exist in a particular accident situation. The physical and chemical forms of the released iodine may change significantly as the result of a change in but one of the many parameters involved. Nevertheless, an understanding of transport and removal phenomena is needed in order to predict the extent of iodine release in a variety of environments as well as the conditions under which it would condense, become adsorbed on particles, react, or remain in the vapor state.

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## 3.3 PARTICLE SIZE AND NUMBER DETERMINATIONS

Radioactive particles generated by overheated reactor fuel in reactor accidents may vary in size from near-molecular dimensions (~20 A in diameter) to sizes large enough to fall rapidly. The latter group will not travel far enough to constitute a hazard outside of the building that houses the reactor. The small particles, however, are essentially unaffected by gravity and may remain airborne for hours or days before they diffuse to a surface and remain there. Such small particles are not easily collected by conventional filtration equipment, and the amounts and types of radioactive fission products associated with them are consequently a matter of concern to hazards analysts. Much effort has, therefore, been devoted to development of methods for characterizing such particles and to collection of information about their behavior in simulated reactor accidents. Techniques for determining sizes and numbers of particles are discussed in this section; collected data are summarized in Sect. 3.4.

### 3.3.1 Diffusion Channel Technique

Diffusion is an important mechanism in aerosol deposition, and, since the diameter of a particle is related to its diffusion coefficient, it is possible to measure the size of particles and to identify molecular vapors by means of their diffusion coefficients. A method of determining diffusion coefficients of fine particles and of radioactive vapors such as iodine by measuring the distribution of radioactivity on the walls of cylindrical and rectangular channels previously exposed to gas carrying radioactive materials and flowing under laminar conditions was described (*49–51*). Since cylindrical tubes are currently being employed exclusively in this work, an equation corresponding to deposition of a given species in a cylindrical diffusion channel, obtained from the equation of Gormley and Kennedy (*52*), was presented (*50*):

$$n_s = \frac{N_0 D}{Q}\left(9.4106e^{-11.489DZ/Q} + 6.8309e^{-70.06DZ/Q} + 5.8198e^{-179.07DZ/Q}\right).$$

In this equation, $n_s$ = number of particles deposited per unit length, $N_0$ = number entering the channel, $D$ = their diffusion coefficient, $Q$ = volumetric flow rate of carrier gas, and $Z$ = distance from channel entrance. For the case of multiplicity of species, the deposition would be given by a summation of expressions of the form of this equation. The diffusion channel method was tested with a narrow (5-mil) rectangular channel and a 26-ml/min flow rate using radioiodine vapor, $^{131}$I-labeled 0.004-$\mu$ aluminum oxide particles, and $^{131}$I-labeled 0.25-$\mu$ tobacco-smoke particles; the particle sizes and/or diffusion coefficients of these three materials were obtainable by independent means. In all three cases, good agreement between theory and experiment was observed, indicating that this method is applicable over the range of interest. The practical upper size limit under conditions currently employed ($^3/_8$-in. tubes, 100-ml/min flow rate) is approximately 100 A (0.01 $\mu$).

Examination of the above equation shows that it describes a straight line only under conditions that permit the second and third terms to be neglected. Calculations (*53*) show that under the conditions listed above, these terms drop out a few centimeters from the entrance of the diffusion tube for particles smaller than 50 A, but for 100-A particles tube lengths greater than 100 cm are required to give a linear deposition rate and different conditions should probably be employed. One fundamental assumption of the diffusion channel method is that the walls are a "perfect sink" for the particles. There seem to be adequate indications in the literature that this assumption is valid for the particle sizes of greatest interest, smaller than 0.1 $\mu$. Wall materials for iodine collection are considered in Sect. 3.5.

### 3.3.2 Filter Analysis Techniques

Silverman and Browning (*54*) have developed a method for characterizing radioactive aerosols by determining their distribution as a function of depth in fibrous filters under carefully controlled conditions. The method distinguishes the contributions of three major processes of filtration: diffusion, interception, and inertial impaction. For experiments with air velocities in the range where diffusion was the controlling mechanism, particle sizes in the range of 40 to 300 A thus determined compared favorably with those obtained from concentration and flow data and from electron photomicrographs. This technique has not been used to date in fission-product release experiments. Browning *et al.* reported (*40*) that the sweep gas from in-pile fuel melting and burning experiments is passed through a series of filters to obtain information about the particulate material which is formed. Figure 3.15 shows the filter assembly which was designed for this purpose. It includes a roughing filter, an HV-70 absolute filter, and a 10-m$\mu$ membrane filter. The sizes of particles collected on the various filters, their states of aggregation, and certain other information about them can be determined by electron microscopy. The gross

PHOTO 64229

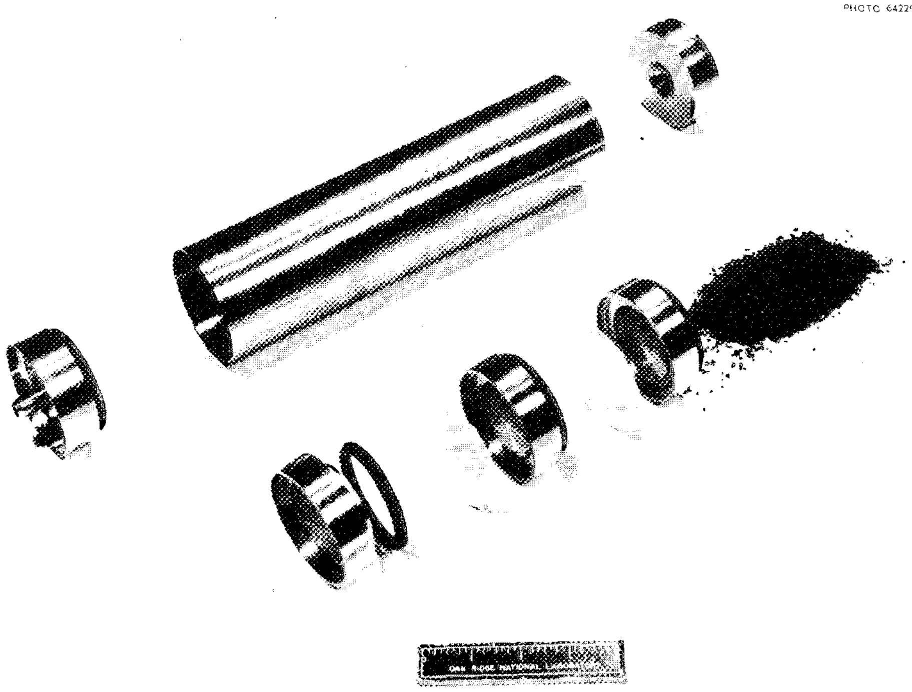

Fig. 3.15. Filter Assembly and Charcoal Trap Used in In-Pile Fuel Destruction Experiments.

amounts of fission products collected are determined by radiochemical analysis.

#### 3.3.3 Deposition Techniques

Both thermal and electrostatic precipitators have been employed to collect small particles generated in simulated reactor accidents. The electrostatic method requires use of a high-voltage device in order to impart a charge to the particles and to permit efficient collection of the charged particles. A device of this type (*55–56*) was employed in the Containment Mockup Facility (*25*) but its efficiency was not evaluated. Two types of thermal precipitators have been described, the hot-wire (*57*) and the hot-plate (*58*) type. Qualitative tests (*57*) indicated a high efficiency for the hot-wire precipitator, and it appears to offer fewer operating problems than the electrostatic precipitator and may be the preferred type for that reason. Samples can be collected in a form suitable for size determinations with an electron microscope. The Andersen Air Sampler (*59*) is useful for collecting particles larger than about 0.6 $\mu$, and it has the advantage that particles in different size ranges are separated and can be submitted for radiochemical analyses or for size evaluations by electron microscope techniques. Other types of inertial samplers have been described (*60*). One that makes use of centrifugal forces up to 26,000*g* has been described by Goetz and Preining (*61*). These investigators state that the aerosol spectrometer, which is commercially available, separates quantitatively airborne particles in the diameter range 3 $\mu$ to 0.03 $\mu$ from the atmosphere in a continuous band-shaped deposit. They applied the instrument to the analysis of natural and artificial aerosols in the submicron range. Craig (*62*) also employed this type of spectrometer to study the interaction of the $Na^{131}I$ vapor with labeled iron oxide particles.

#### 3.3.4 Counting Techniques

The number concentration of particles (population) in aerosols resulting from the overheating of reactor fuels is of interest, and several techniques have been used or are available for use for particle counting. These range from a simple, relatively inexpensive instrument (*63*) for manual counting of the total number of condensation nuclei to more elaborate and more expensive instruments (*64*) that provide counts of the number of particles of various sizes ranging from 0.3 to 16 $\mu$ in diameter. The condensation-type instrument has a maximum scale reading of $10^7$ nuclei/cm$^3$, and it permits size discrimination to a limited degree. The latter instruments use the light-scattering principle, and they have counting rates up to 30,000 particles/min.

### 3.4 SIZE AND ACTIVITY DISTRIBUTION OF PARTICLES FROM OVERHEATED FUELS

Information on the size and, to a lesser extent, the shape and density of particles evolved from overheated fuels in expected reactor environments is needed in order to estimate the hazard of reactor accidents. Aerosols having a high concentration of small particles are highly unstable, and rapid agglomeration occurs until the number concentration is reduced to such a degree that collisions of particles become infrequent. Both theoretical and experimental studies of particle agglomeration have been made, but the effect of radioactivity has not been determined to date at the high specific activity levels that would result from nuclear accidents. Furthermore, the agglomeration process results in large particles with irregular shapes and, in many cases, particles having unknown amounts of void space within them. Methods of relating microscopic observations of particle agglomerates to probable deposition rates apparently remain to be developed. Experimentally measured particle sizes are reported in this section, along with some information on amounts of iodine and tellurium associated with particles of different sizes. Tellurium is of interest to this study because of the $^{132}I$ daughter of 77-hr $^{132}Te$.

#### 3.4.1 Oxidation of Uranium in Air

The temperature at which uranium metal oxidizes in air was found to have a pronounced effect on the size of particles transported (*65*). Less than 4% of the oxide formed at temperatures ranging from 400 to 1200°C was in the size range below 10 $\mu$. At air velocities of 8.3 cm/sec and lower, the quantity of oxide particles, primarily agglomerates of particles 0.015 to 0.5 $\mu$ in diameter, collected 20 in. from the sample, was greatest at 1200°C;

at 400 to 800°C, the average diameter of particles collected under these conditions was 2 to 5 μ. Assuming that the smaller particles, found at both 1000 and 1200°C, arise from the condensation of sublimed $UO_3$, one would predict a large decrease in their concentration with decreasing temperature, as observed. As air velocities are increased, aerodynamic entrainment of larger particles overshadows release by direct volatilization. Rapid cooling of oxidized specimens resulted in projection of microscopic and macroscopic particles at high velocities. Qualitatively, samples with higher burnup were found to exhibit an accelerated particle release (*66*). Results of preliminary studies (*67*) showed that $^{132}Te$ was carried on particles with diameters greater than 200 A, while the majority of $^{131}I$ was carried on particles having a bimodal size distribution with peaks at 15 and 60 A (*67*). In other studies (*68*), also with metallic uranium heated in air, less than 2% of the Te, Cs, Zr, and Ba was found to be associated with particles larger than 1 μ in mean diameter, while 6% of the I deposited with particles larger than 1 μ (*68*). It was reported (*69–70*) that agglomerates of 1- to 2-μ particles were formed when uranium was oxidized at a temperature of 1800°C or higher.

### 3.4.2 Particles Formed from Vaporization of $UO_2$

Significant fractions (up to 25%) of small $UO_2$ specimens vaporized when melted in helium, air, and $CO_2$, although most of the vaporized material deposited in hot regions (*71*). Most of the particles formed were in the 0.01- to 0.1-μ size range, as shown in Fig. 3.16. Iodine, Te, Cs, and Ru were the major fission products transported out of the heated regions by air. Stainless steel cladding altered the particle size distribution (*37*), as shown in Fig. 3.17. When Zircaloy-clad $UO_2$ was melted under the same conditions, the amount of uranium collected on filters was reduced by a factor of 800 as compared with the stainless-steel-clad fuel. Eighty-seven percent of the particles transported from unclad $UO_2$ in out-of-pile tests were less than 0.5 μ in diameter, but when stainless-steel-clad $UO_2$ was heated in-pile with a helium atmosphere, two size groups were found: one centered about 22 A, the other about 30 A (*39*). Thirty-nine percent of the iodine carried out of the furnace was attached to the 22-A particles. Some cesium, strontium, tellurium, and barium were also found on these particles. When a relatively large quantity (35 g) of unirradiated $UO_2$ was maintained above its melting point for 5 min in a helium atmosphere, 10% of the $UO_2$ vaporized but approximately 9%

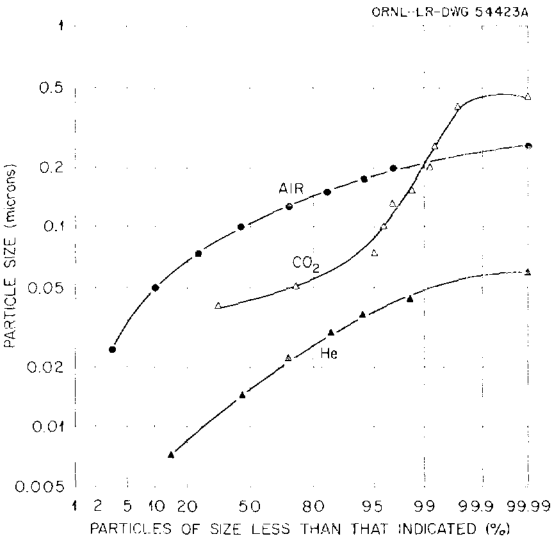


**Fig. 3.16. Particle-Size Evaluation of Oxides Vaporized from Melted $UO_2$ (Arc-Image Furnace).**

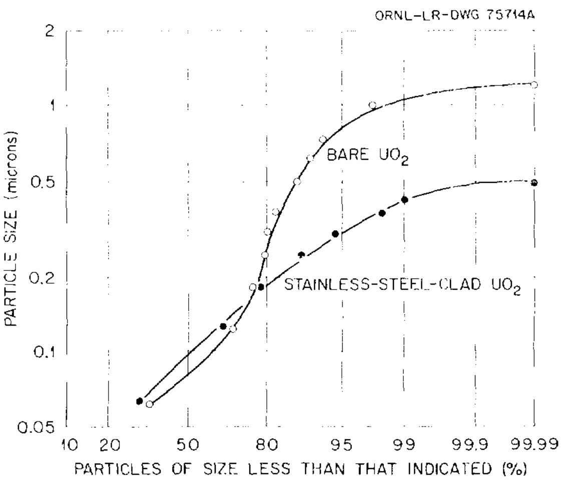


**Fig. 3.17. Comparison of Size Distribution of Particles from Bare $UO_2$ and Stainless-Steel-Clad $UO_2$.**

deposited in the furnace tube and most of the remaining 1% stopped in a low-velocity region. Only about 0.005% reached the filters.

The distribution of released fission products and uranium deposited from a gas flowing laminarly through a diffusion tube was analyzed to yield information about vapors, gas-borne particles in the range of 10 to 100 A in diameter, and the amounts of released material associated with these forms (*41*). In most of the experiments in which stainless-steel-clad $UO_2$ specimens were heated in-pile, the data indicate that particles having an average diameter of about 35 A were released. Considerable amounts of released $^{131}I$, $^{132}Te$, and $^{137}Cs$ are always carried by these fine particles. Particle size distribution data obtained from filters in two of these experiments are shown in Fig. 3.18, along with some data on particles from uranium carbide burning experiments discussed in the following section. Similar data obtained in experiments (*72*) to determine fission-product release from stainless-steel-clad $UO_2$ under transient reactor conditions are illustrated by Fig. 3.19. The data indicate that an oxidizing atmosphere resulted in a larger number of small particles than were produced (*73*) in helium (experiments 1 and 2).

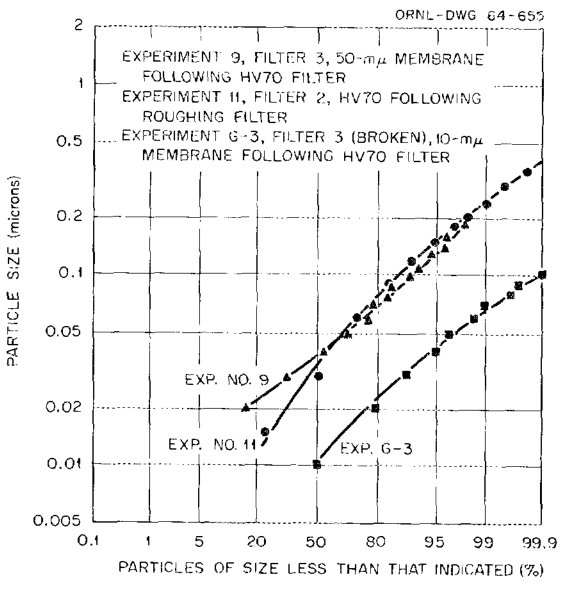


**Fig. 3.18. Size Distribution of Particles Retained on Filters in In-Pile Fuel Destruction Experiments.**

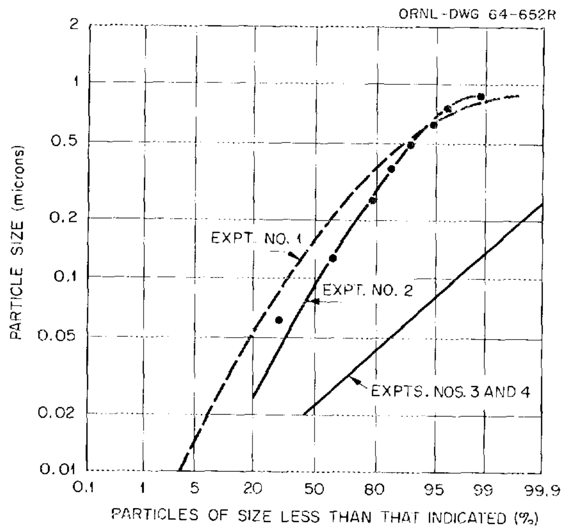


**Fig. 3.19. Sizes of Particles Collected on Filters During In-Pile Melting of Stainless-Steel-Clad $UO_2$ Under Transient Reactor Conditions.**

### 3.4.3 Particles Formed by Oxidation of Uranium Carbide

Little information seems to be available on particle distribution resulting from exposure of uranium carbide to oxidizing atmospheres. Two experiments have been reported (*40*) in which fuel specimens composed of pyrolytic-carbon-coated uranium carbide particles in a graphite matrix were partially destroyed by exposure to air in the ORR at temperatures ranging from about 890 to 1400°C. Varying amounts of iodine (13 or 58%) and of tellurium (8 or 32%) were found to be associated with particles in the 15- to 30-A size range. The size distribution of particles collected on two filters in series during one of these experiments is shown in Fig. 3.20. The median particle diameter of the particles collected on filter No. 2 (type E glass) was 1.2 μ, while that on filter No. 3 (HV-70) was 5 μ. In another experiment of this type, particles collected on a 10-mμ membrane filter after passing through an HV-70 filter showed a median diameter of 0.01 μ, but this filter was broken, possibly during assembly of the filter holder, so there is doubt as to the validity of the data. Size distribution data obtained from this filter are included in Fig. 3.18.

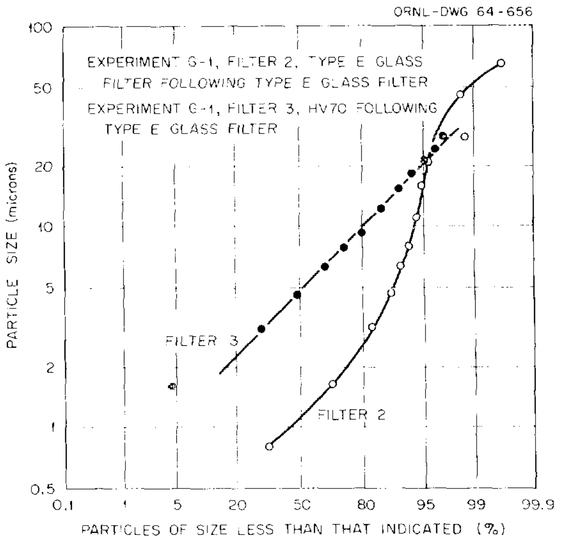


**Fig. 3.20. Size Distribution of Particles Retained on Filters in Fueled Graphite Burning Experiments.**

### 3.4.4 Particles Produced on Melting Irradiated Aluminum-Uranium Alloys

Some observations were made (*25*) of the particles produced when highly irradiated (23.6% burnup) aluminum-uranium alloy specimens were melted in a steam-air mixture at about 750°C. Particles remaining airborne in the CMF tank for 1 to 2 hr were collected by an electrostatic precipitator and examined by electron microscopy. The smallest particles noted were approximately 150 A in diameter; the largest had an average diameter of about 1500 A. A portion of the aerosol was simultaneously passed through a 10-ft length of silver-coated diffusion tube, and the remaining particles collected on a membrane filter were found to range in diameter from about 0.05 to 3.0 $\mu$. The large particles were agglomerates of small particles.

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## 3.6 ORGANIC IODINE COMPOUNDS

Radioiodine combined in a form now believed to be organic iodine compounds has been observed in the aerosol formed by heating both lightly irradiated and highly irradiated uranium and in that produced by use of a radioisotope source under a variety of conditions (*45*). The fraction of radioiodine of this form observed in various experiments has varied from a few tenths of a percent to ~30%. Elemental iodine is comparatively easily removed from gas streams, but the organic iodine compounds, being less reactive, are more difficult to remove and activated charcoal is the only material presently known to be capable of trapping them with a reasonable degree of efficiency. A knowledge of the identity of such compounds and of their behavior would be of considerable value in devising better iodine trapping systems and in predicting the probability of escape of iodine in that form in reactor accidents. It should be recognized that the biological half-life of these compounds has not been determined but it seems likely to be much shorter than that of elemental iodine. Establishment of a half-life value for iodine in this form may materially reduce requirements for their retention by reactor containment systems. Methods employed (*45*) for the separation and identification of the organic iodine compounds are discussed in the following sections, along with some information on their chemical behavior in aqueous systems and in a May Pack.

### 3.6.1 Solution Studies

Atkins and Eggleton (*45*) found evidence of two distinct types of iodine compounds (designated fractions A and B) present in an aerosol formed by oxidation of carrier-free $Na^{131}I$ at 400°C in air purified by passage through a charcoal trap at −78°C. Elemental iodine and particulate matter were first removed from the effluent air stream, and it was then bubbled through dilute caustic or sulfuric acid solutions. About 95% of the iodine activity was removed from the air stream by either reagent, mostly in the first bubbler in the train.

When the bubbler solutions were equilibrated with benzene, fraction A extracted into the organic phase with a high partition coefficient and was not back-extracted with NaOH solution. Fraction B, which was apparently more reactive than fraction A, extracted into benzene with a partition coefficient between 1 and 5 and reacted irreversibly with NaOH to give a form insoluble in benzene. Approximately equal quantities of fractions A and B were found in the bubblers.

Oxidation of the mixed fractions with ceric sulfate resulted in the loss of benzene solubility.

Sulfur dioxide reduction, with added iodide carrier, followed by oxidation of the carrier to iodine, gave a form of the compound insoluble in benzene. Coprecipitation studies showed that approximately 5% of the mixed activities was carried on the precipitate in the stepwise precipitation of silver iodate. The mixed activities were carried quantitatively on silver iodide following reduction with sulfur dioxide.

### 3.6.2 Gas-Liquid Chromatography

A gas chromatography apparatus incorporating both a katharometer (apparatus for measuring thermal conductivity of gases) and a radioactivity detector (gamma-scintillation counter) was used to identify the compounds in fraction A. Columns 2 m long and of 6 mm internal diameter containing silicone oil or dinonyl phthalate on Celite as the stationary phase were used with helium as carrier gas. Flow rates of approximately 50 $cm^3$/min were maintained with column temperatures of 40 ± 1°C. Samples were collected by passing the effluent air stream from the iodine generator through a liquid-oxygen-cooled trap after removal of molecular iodine, particulate material, and fraction B, the last by passing the air through a magnesium perchlorate drying tube. Fraction B has apparently not been studied.

Examination of fraction A showed it to consist of several compounds. The radioactivity detector gave evidence for seven compounds, with the radioactivity in the first (peak 1) comprising about 90% of the total. Since the retention volume for stable methyl iodide obtained using the katharometer was found to be the same as that of peak 1, the identity of this compound was confirmed as methyl iodide.

Table 3.3 contains the measured retention volumes for peaks 1–7, together with those obtained for the series of alkyl iodides arranged in order of their boiling points. The good agreement obtained should not be regarded as final confirmation of the identity of peaks 2–7, but taken in conjunction with the known identity of peak 1, the evidence strongly suggests that these substances are the higher alkyl iodides shown in the table.

Oxidation of lightly irradiated uranium gave rise to a compound having the same retention volume on a silicone oil column as methyl iodide. A more accurate comparison of the retention volume of this compound with that for methyl iodide was made by a new method, as follows. A mixture of the radioactive species and an inactive sample of methyl iodide was passed onto the chromatography column, and the peak as indicated by the katharometer was split into two approximately equal parts, each of which was collected in a liquid-nitrogen-cooled trap. The masses of the approximately equal parts were determined from the peak areas obtained on passing each part separately through the column again. The peaks on emerging from the column were absorbed in small charcoal-loaded traps, and their activities were measured.

**Table 3.3. Retention Volume (Uncorrected for System Dead Space) for Simple Alkyl Iodides and for Peaks 1–7 of Fraction A.** Silicone oil (15 wt %) on Celite. Temperature 40 ± 1°C. Flow rate 48 $cm^3$/min (*45*)

| Material | Boiling Point (°C) | Retention Volume, $V_R$ | Peak from Organic Fraction | $V_R$ |
|---|---|---|---|---|
| Methyl iodide | 42.5 | 214 | 1 | 214 |
| Ethyl iodide | 72.2 | 424 | 2 | 422 |
| Isopropyl iodide | 89.5 | 690 | 3 | 680 |
| *n*-Propyl iodide | 102.4 | 941 | 4 | 897 |
| *tert*-Butyl iodide | 100 (d) | 962 | 5 | 979 |
| *sec*-Butyl iodide | 117.5 | 1614 | 6 | 1750 |
| *n*-Butyl iodide | 131 | 2370 | 7 | 2430 |

From these figures the specific activity of each part was calculated. These were found to agree within 1%. The column had an efficiency of approximately 1000 theoretical plates for methyl iodide. Calculation shows that for a column of this efficiency, a difference in the retention volumes of the active and inactive species of only 0.1% would lead to a difference of more than 5% in the specific activities of the two halves. Consequently, the identity of this compound as methyl iodide was confirmed.

### 3.6.3 Molecular Weight Determination

The molecular weight of the major constituent of fraction A was determined by a modification of the classical effusion method. For the gas effusing through a hole whose diameter is small compared with the mean free path of the gas, the rate is proportional to the pressure and inversely proportional to the square root of the molecular weight. In the apparatus constructed the gas was allowed to effuse through a small hole in a platinum diaphragm mounted in the stopcock of a flask into a high vacuum. The flask was placed on the crystal of a gamma-scintillation counter, and the fraction of the radioactivity remaining after various effusion times was measured. The amount remaining decayed exponentially with time, the half-life being proportional to the square root of the molecular weight. A calibration curve was obtained for the apparatus with $^{85}Kr$ and $^{133}Xe$. Using this calibration curve, the measured value of the molecular weight of peak 1 in fraction A was found to be 152. Pure methyl iodide was labeled with $^{131}I$ by an exchange technique, and the measured value of its molecular weight was found to be 150 (theoretical value 146), thus confirming the identity of peak 1 as methyl iodide.

### 3.6.4 Retention of Methyl Iodide by May Pack Components

Following the identification of methyl iodide as the major constituent of fraction A, Atkins and Eggleton (*45*) studied the behavior of pure methyl iodide labeled with $^{131}I$ in a May Pack. The results, summarized in Table 3.4, show that, with a flow rate of 10 liters/min, all the methyl iodide was retained by the pack, with 92% of it being collected by the charcoal bed.

**Table 3.4. Retention of Methyl Iodide on May Pack Components (*45*)**

| May Pack Component (in order of assembly) | % Retention of Methyl Iodide |
|---|---|
| 1. Copper gauze 1 | 1.5 |
| 2. Copper gauze 2 | 1.7 |
| 3. Wiggins Teape charcoal paper | 1.2 |
| 4. Whatman ACG/B charcoal paper | 3.6 |
| 5. Charcoal pack (20 g) | 92.0 |

## 3.7 WET CHEMICAL METHODS OF IODINE FRACTIONATION

The distribution of iodine between water and organic solvents and other wet chemical tests have been used to determine valence states of iodine. This technique has not been widely applied in fission-product-release experiments. Two investigations of this type are included here.

### 3.7.1 The Chemical State of Iodine Released into Steam (*78*)

The chemical states of iodine released from metallic uranium specimens irradiated over the range $2.0 \times 10^{14}$ to $5.4 \times 10^{19}$ *nvt* and from $UO_2$ powder irradiated up to $2.0 \times 10^{16}$ *nvt* were studied with fuel temperatures in the range 1000 to 1350°C, using the apparatus shown in Fig. 3.26.

In this apparatus, steam passes over the inductively heated fuel sample and carries released fission products into the condenser. At the completion of an experiment, the boiler and condenser aqueous solution and the bubbler solutions are individually analyzed using wet chemical procedures. The distribution of iodine in the various traps depends on the boiling and reflux rates within the heated cell; however, in general, more than 90% of the released iodine is collected along with the condensed steam in the condenser flask.

Typical results for the release of iodine from metallic uranium are listed in the second column of Table 3.5. The majority of the iodine (av 84%) was found in the reduced state, while the remainder was primarily elemental; less than 1% was in the iodate and periodate states. Essentially no difference in the iodine chemical states was found by varying the pH of the water initially charged to the cell between 5.9 and 6.8. Successive filtrations through 1.2-$\mu$, 0.45-$\mu$, 0.3-$\mu$, 100-m$\mu$, and 10-m$\mu$ Millipore filters showed that none of the iodine was associated with particles larger than 10 m$\mu$.

In contrast to the metallic uranium experiments, an average of 80% of the iodine released from $UO_2$ into steam was found to be elemental. These results are shown in the third column of Table 3.5. Filtration experiments similar to those mentioned above also showed that none of this iodine was associated with particles larger than 10 m$\mu$, but dialysis membrane retention studies indicated that up to 8% of the iodine may have been attached to smaller particles.

**Table 3.5. Average Percentages of Chemical States of Iodine**

| State | Fraction in Each State (%) | |
|---|---|---|
| | $U\text{-}H_2O$ System | $UO_2\text{-}H_2O$ System |
| $I^0$ | 15 | 80 |
| $I^-$ | 84 | 10 |
| $I^{5+,\ 7+}$ | <1 | <0.5 |
| Other | 1 | 10 |

During the course of an experiment the heated uranium metal reacts with the steam to form $UO_2$ and evolve $H_2$ and fission products. The initial reaction rate is rapid, and a plot of rate vs time is a parabolic curve. This is followed by a linear reaction rate, during which time an oxide layer of constant thickness is maintained on the surface.

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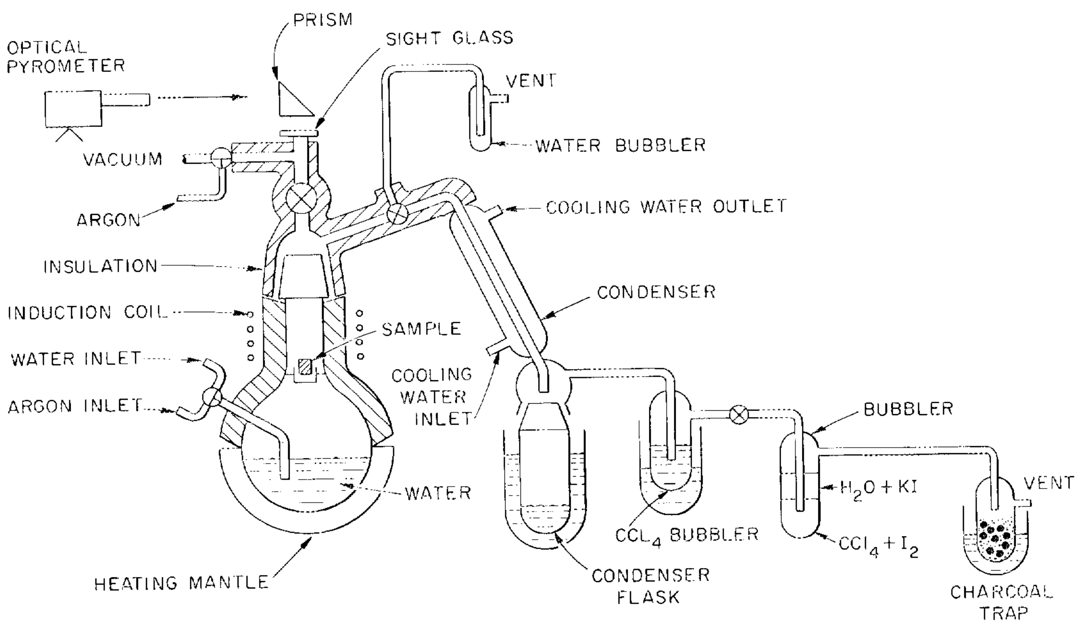


**Fig. 3.26. Apparatus for Determining the Chemical Form of Iodine Released from Irradiated Fuel in a Steam Atmosphere.**

The release from the $UO_2$ powder samples is primarily by diffusion, perhaps enhanced by the lattice expansion resulting from the conversion to $UO_{2.2}$, the thermodynamically stable oxide under the prevailing conditions. Only a small quantity of $H_2$ is formed by this oxidation process; this is added to the small quantity resulting from the dissociation of steam at the fuel temperatures. Basically, then, the iodine is released along with other fission products from a $UO_2$ surface in both the metallic uranium and $UO_2$ studies, the major difference being the quantity of $H_2$ in the atmosphere surrounding the fuel. Most of the iodine is dissociated to atomic iodine (I) at the temperatures used (1000 to 1350°C). Although atomic I is quite reactive because of its unpaired electron, highly exothermic gas phase reactions with atoms of other fission products to form simple molecules are precluded, since the energy of reaction goes to a vibrational mode and the molecules dissociate. Third-body collisions are required to stabilize the products. Reactions with $H_2$ should proceed, however, since one hydrogen atom removes the energy of reaction in the form of the translational energy.

The importance of reactions between $H_2$ (resulting from a metal-water reaction) and iodine was confirmed by introducing $H_2$ into the release cell during a $UO_2$ release experiment. The final iodine chemical states were similar to those observed in experiments with metallic uranium, and again most of the iodine was in the reduced state. The important reactions are shown below.

$$2H_2O + U(F.P) \rightarrow UO_2 + 2H_2 + F.P. \,, \qquad (1)$$

$$H_2 + 2I \rightleftharpoons 2HI \,, \qquad (2)$$

$$2I \rightleftharpoons I_2 \,, \qquad (3)$$

$$2I + H_2O \rightleftharpoons HIO + H^+ + I^- \,, \qquad (4)$$

$$3IO^- \overset{\Delta}{\rightleftharpoons} IO_3^- + 2I^- \,. \qquad (5)$$

The fission products are released by reaction (1). The released iodine reacts with the $H_2$ to form HI by (2). The equilibrium expressed in reaction (3) is far to the left, so that little molecular iodine is present within the cell. A small quantity of iodine reacts according to equation (4), but the hydrolysis is very small (see Sect. 2.2.1). Some disproportionation of $IO^-$ takes place at elevated temperatures in solution, giving rise to iodates and periodates via reaction (5). These latter species account for less than 1% of the iodine. Some of the iodine (about 1% from uranium and up to 10% from $UO_2$) is not in any of the above-mentioned states and is probably in the form of organic iodides. The origin of these species is under study.

The equilibrium between $H_2$, I, and HI was calculated for the iodine concentration range $10^{-8}$ to $10^{-14}$ mole/liter, $H_2$ partial pressures ranging from 1 to $10^{-5}$ atm, and temperatures between 298 and 2000°K. The results of these calculations are shown in Fig. 3.27. In the experiments reported here, the bulk average cell temperature was between 500 and 1000°K and the $H_2$ partial pressure was approximately $10^{-3}$ atm. Referring to Fig. 3.27, from 75 to 95% of the iodine should be converted to HI under these conditions. All of the metallic uranium experimental results to date fall within this range.

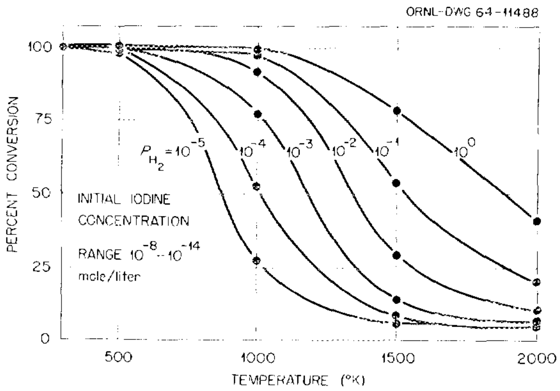


Fig. 3.27. Hydrogen Iodide Equilibrium.

### 3.7.2 Solubility Behavior of Volatile Iodine Compounds (*79*)

It has been demonstrated by diffusion tube analysis that iodine sources nominally in the elemental form can have a small fraction of the iodine existing in compound forms which are rather difficult to remove from air under certain conditions. At least three vapor species have been distinguished, and workers in England (*45*) have identified one of the forms of gas-borne radioiodine as methyl iodide. Figure 3.28 illustrates the apparatus used to perform an experiment to determine whether methyl iodide is contained in the A fraction of iodine sources prepared by the palladium iodide method

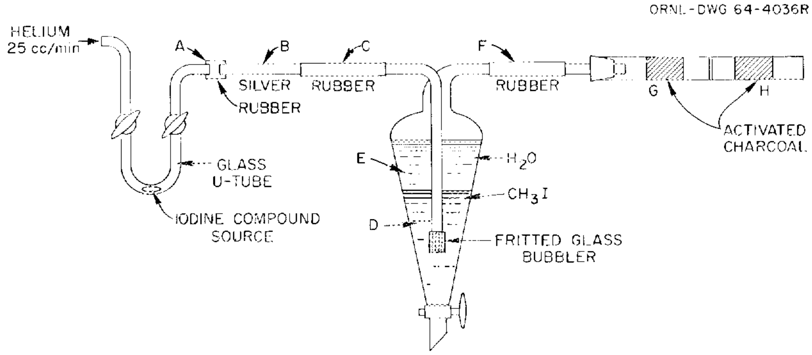


**Fig. 3.28. Apparatus for Investigating Solubility of Selected Iodine Vapor Species.**

and also to investigate the applicability of differential solubility as a tool to characterize the volatile iodine compounds. Three experiments were performed. In the first two experiments the A fractions were separated from iodine sources by use of different techniques, were placed in independent containers, and were introduced into the experiments. In the third, pure methyl iodide vapor, tagged with radioactive methyl iodide (tagged methyl iodide obtained from Volk Radiochemical Co., Skokie, Ill.; produced by the exchange of $CH_3Br$ and $Na^{131}I$ in acteone), was introduced into the experiment. A gas sweep, containing the volatile iodide compound, was passed through a silver-lined tube and a rubber tube prior to entering the separatory funnel containing nonradioactive methyl iodide and water. In the first two experiments, the distribution coefficient between the two phases was about 55 in favor of methyl iodide. This corresponds approximately to the expected distribution considering the solubility of methyl iodide in water. The distribution coefficient in the third experiment using tagged methyl iodide was comparable. The results, presented in Table 3.6, indicate that the mixture of iodine vapor species emanating from the iodine vapor source includes methyl iodide or an iodine compound of similar solubility.

Table 3.6. Sorption and Solubility Behavior of Volatile Iodine Compounds

| Sample Identity (see Fig. 3.28) | Relative Amount of Activity[a] | | |
|---|---|---|---|
| | Exp. 1[b] | Exp. 2[b] | Exp. 3[c] |
| A – rubber tube | 0.39 | 0.019 | 0.008 |
| B – silver tube | 0.92 | 0.023 | 0.001 |
| C – rubber tube | 0.69 | 0.32 | 0.92 |
| D – methyl iodide | 1.00 | 1.00 | 1.00 |
| E – water | 0.02 | 0.015 | 0.017 |
| F – rubber tube | 0.001 | 0.014 | 0.011 |
| G – charcoal | 0.008 | 0.053 | 0.073 |
| H – charcoal | 0.007 | 0.003 | 0.000 |
| Distribution Coefficient ($CH_3I/H_2O$) | | | |
| | 49.6 | 61.9 | 46.2 |

[a]Data in each column normalized to methyl iodide sample = 1.00.

[b]Experiments using A fraction consisting of iodine compounds volatilized at −70°C.

[c]Experiment using $^{131}I$-labeled $CH_3I$.

# 4. TRANSPORT OF FISSION-PRODUCT IODINE AND NATURAL REMOVAL PHENOMENA

Natural phenomena affecting the transport behavior of released fission-product iodine are considered in this section. Induced removal and trapping methods are treated in Sect. 5.

A great deal of effort has been directed to the problem of defining the probable behavior of iodine in reactor accidents. Although some allowance is made (*4–5*) for deposition of iodine in a containment shell, it is generally believed that the deposition factor (0.5) employed in TID-14844 and in many hazards summary reports may be quite conservative and that a much higher attenuation factor could be justified by availability of more complete information on iodine deposition rates in containment systems. The complex chemical behavior of iodine demonstrated in the previous section and the various uncertainties involved in defining accident conditions combine to make prediction of the extent of iodine deposition such a difficult problem that it may be necessary to accept a considerable degree of uncertainty in the iodine deposition factor. Because of this fact, engineered safeguards discussed in Sect. 5 may be required to supplement iodine attenuation in containment systems by natural removal methods. Available information, including some data obtained under conditions that are unlikely to prevail in reactor accidents, is summarized here in an effort to provide a reasonably clear picture of the present state of knowledge in this field rather than final answers to the problems involved.

## 4.1 TRANSPORT ON PARTICLES

Following its release from the fuel, fission-product iodine will be transported in the primary system and then into the containment volume. Part of the released iodine will deposit in close proximity to the fuel and is of little consequence to the overall hazard. The remainder, which may be in the vapor state or attached to particulate matter, is transported out of the high-temperature region, where it may condense, agglomerate and settle, or, in the case of volatile vapors and fine particles, remain airborne for extended periods of time. The ultimate deposition of the iodine depends not only on its chemical and physical properties but also on the properties of the particles to which it may become attached.

High concentrations of vaporized fuel particles in conjunction with the static gas surrounding the heated fuel element may result in formation of large particles which will settle out in a reactor pressure vessel and carry much of the released iodine with them.

### 4.1.1 Sorption on Particles

As Chamberlain (*81*) has pointed out, particles may remove iodine from the atmosphere by physical adsorption, a reversible process, or by chemisorption, which is irreversible (at least at normal room temperatures). He adapted Fuchs' equation (*82*) for evaporation from small droplets to give the following equation for the rate of removal of iodine from the atmosphere by the irreversible process:

$$\frac{1}{C}\frac{dC}{dt} = \frac{-4\pi r n D}{(D/rN\alpha) + (r/r + \Lambda)}, \qquad (1)$$

where

$t$ = time, sec,

$C$ = concentration of iodine in air,

$r$ = radius of particles, cm,

$n$ = number of particles per cm$^3$,

$D$ = diffusion coefficient of iodine in air,

$R$ = gas constant,

$T$ = absolute temperature,

$M$ = molecular weight of iodine,

$N$ = $(RT/2\pi M)^{1/2}$ ($N$ has the value 3913 cm/sec at 20°C for $I_2$),

$\alpha$ = accommodation coefficient or sticking probability of iodine on particles,

$\Lambda$ = mean free path of iodine molecules in air ($= 4.19 \times 10^{-6}$ cm at 20°C).

For very small particles ($r << \Delta$), this equation reduces to:

$$\frac{1}{C}\frac{dC}{dt} = -4\pi r^2 nN\alpha \ . \qquad (2)$$

For particles of about micron size, $r >> \Delta$ and $r/r + \Delta = 1$. The first and second term in the denominator on the right hand side of the more general equation above may then be dominant, depending on whether $D/rN\alpha$ is greater or less than unity. If $r = 10^{-4}$ cm, $D = 0.080$ cm$^2$/sec, and $N = 3913$ cm/sec, the condition $D/rN\alpha > 1$ is equivalent to $\alpha < 0.2$. Thus, if the accommodation coefficient is less than 0.2, its magnitude will determine the rate of adsorption of iodine. For 10-$\mu$ particles $\alpha$ must be less than about 0.02 for the same to be true, and in general the larger the dimensions of the surface the less important is $\alpha$ in determining the rate of uptake.

In Table 4.1 is shown (*81*) the theoretical rate of uptake of iodine for various aerosols according to Eq. (1), with values of the half-life in the gas phase calculated for $\alpha = 1$ and for $\alpha = 10^{-4}$.

In practice, the value of $\alpha$ is likely to depend on the nature of the surface of the particle, and in particular on the amount of iodine adsorbed there. Also shown in Table 4.1 is the amount of iodine in $\mu$g per m$^3$ of air which would give a monolayer coverage on the particulate aerosols.

Megaw and May (*83*) estimated accommodation coefficients of $4.7 \times 10^{-3}$ and $1.2 \times 10^{-3}$ for iodine adsorption on Aitken nuclei in two tests performed in the Pluto reactor shell. They state that in small-particle studies accommodation coefficients of unity are generally assumed, although Chamberlain states (*81*) that chemisorption on such particles has not been observed. Consequently, they interpreted the low values obtained in these experiments, together with the observation that radioiodine adsorbed on small particles did not appear to interchange with inactive iodine, as indicating that irreversible chemical adsorption of iodine occurred with only a small fraction of the nuclei present in the gas. This theory is also supported by data obtained in the same experiments showing that the fraction of iodine attached to particles dropped off more rapidly than did the number of Aitken nuclei. The authors state that the data available did not permit a decision as to whether the particulate fraction of the iodine is the result of attachment of iodine atoms to a relatively small proportion of chemically suitable nuclei or the result of attachment to an even smaller number of much larger particles. They favored the first alternative.

**Table 4.1. Theoretical Rate of Uptake of Iodine on Particles in Air (*81*)**

| | Aerosol Type and Number | | | | |
|---|---|---|---|---|---|
| | Aitken Nuclei | | Lead Fume | Water Fog | Raindrops |
| | I | II | III | IV | V |
| Particle radius, $\mu$ | 0.01 | 0.1 | 1 | 10 | 500 |
| Particle density | 3 | 3 | 9 | 1 | 1 |
| No. per cm$^3$ | $10^5$ | $10^4$ | $10^3$ | $10^2$ | $10^{-4}$ |
| Terminal velocity, cm/sec | $4 \times 10^{-5}$ | $6 \times 10^{-4}$ | $7 \times 10^{-2}$ | 1.2 | 400 |
| Mass per m$^3$ of air, mg/m$^3$ | $1.2 \times 10^{-4}$ | $1.2 \times 10^{-2}$ | 38 | 420 | 52 |
| Surface area per m$^3$ of air, cm$^2$/m$^3$ | 1.2 | 12 | 120 | 1200 | 3.1 |
| Mass of $I_2$ to give monolayer, $\mu$g/m$^3$ | 0.4 | 4 | 40 | | |
| Half-life of $I_2$ in gas phase, min | | | | | |
| $\alpha = 1$ | 2.4 | 0.33 | 0.13 | 0.12 | 480[a] |
| $\alpha = 10^{-4}$ | $2.4 \times 10^4$ | 2400 | 240 | 24 | 2000[a] |

[a]These figures include allowance for the "ventilation factor" which is the increased rate of transfer caused by the fact that large droplets move rapidly through air.

Chamberlain (*81*) reported that, in experiments performed by Eggleton and Cousins, when uranium oxide particles with an effective diameter of 0.3 $\mu$ (7 mg/m$^3$) produced by burning uranium were introduced into a box containing 0.09 $\mu$g of iodine vapor per m$^3$, equilibrium was established in less than 10 min when about half of the iodine was adsorbed on particles. When air was passed through a filter paper on which particles were collected, most of the iodine was removed, indicating reversible adsorption. Experiments with "Magnox" oxide (Magnox contains about 99% Mg and 1% Al) were said (*81*) to yield confusing results, with the fraction of the total iodine adsorbed reversibly varying erratically. A maximum adsorption of 1.5 $\mu$g of $I_2$ per mg of oxide was achieved, corresponding to about 0.1% of a monolayer coverage. A drastic reduction in rate of deposition of iodine on silver and other surfaces after the introduction of MgO particles was noted. In other experiments (*81*), also performed with Magnox oxide, the amount of iodine adsorbed on the oxide particles corresponded to only 1% of a monolayer with iodine concentrations (in air) of 5 $\mu$g/m$^3$ and 13 mg/m$^3$. The iodine appeared to be irreversibly adsorbed on the oxide at both iodine concentrations. The uptake of iodine on Magnox wire was found to be roughly proportional to the iodine concentration in the gas phase in the range 40 $\mu$g/m$^3$ to 60 mg/m$^3$, with more than a monolayer amount adsorbed at the latter concentration. The reason for the different behavior of the two materials might be accounted for by the fact that the wire had an MgO surface over a metal substrate, implying that the iodine vapor may penetrate the oxide coating and react with the metal.

Chamberlain also points out (*81*) that normal methods of sampling for iodine do not permit determination of the amount of iodine adsorbed reversibly on particles and that this becomes confused with the radioiodine vapor fraction or the iodine compound fraction, depending on the type of sampler used. He also states that both reversible and irreversible adsorption of iodine on particles will result in a much lower rate of deposition than for iodine vapor and that it would therefore be unwise to consider the rapid deposition of iodine observed in the Dido-Pluto experiments to be typical of all types of reactor accidents.

Craig (*62*) has summarized available information on interaction of particles and airborne radioactive materials, and he also reported the results of an investigation of the adsorption of sodium iodide vapor containing $^{131}$I on iron oxide particles labeled with $^{59}$Fe. He used the Goetz Aerosol Spectrometer to characterize the adsorption of NaI molecules on the iron oxide aerosol as a function of particle size with radionuclide-to-aerosol concentration ratios of $3 \times 10^2$ to $3 \times 10^5$ molecules per particle and with mean interaction times varying by a factor of 2.5. His conclusions were: (1) considerable quantities of the radionuclide became attached to the aerosol but the adsorption process was not irreversible; (2) variation of the experiment conditions over the above-mentioned range did not significantly affect the results; and (3) the adsorbed activity was proportional to a function $D(I)^Z$, where $D(I)$ is the aerosol diameter and $2 < Z < 3$, over the approximate size range 0.06 to 0.65 $\mu$ and possibly up to particle diameters of 1.2 $\mu$.

The first conclusion is rather surprising considering the fact that NaI (reported boiling point 1300°C) undoubtedly has a very low vapor pressure at room temperature. The author apparently failed to examine the possibility that impurities in the stream of argon that was used to transfer the radionuclide vapor from the furnace where NaI was heated into the aerosol chamber might have produced some molecular iodine vapor or other volatile iodine compounds. Therefore, his conclusion concerning the reversible adsorption of NaI molecules on iron oxide particles seems questionable.

### 4.1.2 Transport on Fuel and/or Cladding Particles

Many experiments on release and transport of fission products from irradiated fuels have been performed, but there seems to be little data in the literature that will permit a valid comparison of transport of iodine liberated from clad and unclad fuel in air or steam-air mixtures. Consequently, no attempt will be made in this section to define the effect of particles of vaporized cladding materials on iodine transport.

Data (*84*) on the amount of iodine liberated by heating Al-U alloy fuel that was carried by particles is given in Table 4.2, as a function of fuel temperature in different atmospheres.

**Table 4.2. Transport of Fission-Product Iodine Released from Aluminum-Uranium Alloy (*84*)**

| Fuel Temperature (°C) | Percent of Iodine Inventory Transported to Filters by Particles | | |
|---|---|---|---|
| | Air | Helium | Steam-Air[a] |
| 700 | 2.8 | | 0.3 |
| 800 | 4.5 | 1.4 | 0.1 |
| 900 | 4.2 | 1.5 | 0.7 |
| 1000 | 0.9 | 13.3 | 4.1 |
| 1100 | 3.9 | 17.4 | 3.9 |
| 1150 | 9.2 | | |

[a]80% steam and 20% air, by volume.

The experiments conducted in helium and in a steam-air mixture show some evidence of increased transport of iodine on particles with increased temperature, but the air experiment data show too great a scatter to demonstrate a definite trend. In one experiment, where the fuel specimen was heated to 1100°C and 17% of the iodine was caught on the particulate filter, the particles were found to vary in size from 0.04 to 4 $\mu$. The deposit on the filter contained zinc (an impurity in the fuel) and aluminum. When a helium stream was passed over heated alloy material in a quartz tube, a shiny deposit on the wall of the tube was observed and was identified as zinc. The deposit contained more than 90% of the released iodine and almost half the cesium but very little uranium. Diffusion tube studies of the aerosol produced in the steam-air experiment, after it had aged for 1 to 2 hr, showed that 95% of the iodine present was in the molecular form, while the other 5% was attached to particles with a calculated diameter of 26 A. These figures correspond to 17% and 10% of the original fuel iodine inventory.

The transport of iodine liberated by heating $UO_2$ fuel, especially stainless-steel-clad specimens, has been examined in both in-pile and out-of-pile experiments. Data (*40*) on the size of small particles produced by heating clad $UO_2$ in flowing helium with fission heat in the Oak Ridge Research Reactor were obtained by using the stainless steel exit tube from the furnace as a diffusion device. The particles with which iodine (as well as other fission products) was associated were found to have an average diameter of about 20 A, and variable amounts of the iodine (3 to 56%, average 23%) were apparently carried by these small particles. Similar results were obtained when pyrolytic-carbon-coated uranium carbide particles in a graphite matrix were partially destroyed by burning in the ORR in-pile facility.

Out-of-pile studies (*85*) of the aerosol produced by melting trace-irradiated, stainless-steel-clad $UO_2$ fuel in air after a 2- to 3-hr aging period showed that 80% of the airborne iodine was in the molecular form, about 15% was attached to particles with an apparent diameter of 26 A, and 5% was associated with particles 108 A or larger in diameter. These values correspond approximately to 20%, 4%, and 1%, respectively, of the original iodine inventory of the fuel.

### 4.1.3 Agglomeration and Settling of Particles

Data indicating that significant quantities of radioactive iodine released in nuclear accidents may be attached to particles are discussed in the previous sections. It becomes of interest to examine the rate at which such particles might disappear from the atmosphere of a reactor containment shell. Stokes' law gives the velocity (in cm/sec) for the fall of small spheres in air as

$$V = \frac{2}{9}\, r^2 \rho g/\eta = 1.2 \times 10^6\, \rho r^2 ,$$

where $r$ is the particle radius in cm, $\rho$ is the particle density, $\eta$ is the coefficient of viscosity of air, and $g$ is the acceleration of gravity. This equation is said (*86*) to hold within 5% for particles in the 1- to 50-$\mu$ range. At the lower end of the scale, the calculated velocity for a particle with a density of 5 (approximate value for iron oxide) is 0.6 mm/sec, while the corresponding velocity for 10-$\mu$ particles is 6 cm/sec. When the aerosol is stirred, as by convection currents, the number concentration $n$ of particles at a given time $t$ is represented by the equation (*86*):

$$n = n_0\, e^{-vt/h} ,$$

where $v$ is the velocity of fall of particles in a rectangular box of height $h$. This equation shows that the number concentration of particles and the rate of settling (in terms of the number of particles per second) decrease exponentially with time.

The differential equation governing the decrease in number of particles due to coagulation is (*87*):

$$\frac{dn}{dt} = Kn^2 .$$

This equation holds only for particles that are large compared with the mean free path $l$ ($10^{-5}$ cm in air at room temperature). For smaller particles, the Cunningham slip correction (*88*) must be applied and the corrected equation is:

$$\frac{dn}{dt} = K(1 + 0.9\ l/r)n^2 .$$

This shows that 0.1-$\mu$ particles coagulate 88% faster than 10-$\mu$ particles.

Since $K$ has the approximate value $5 \times 10^{-10}$, it is obvious that coagulation rates are likely to be low for particle concentrations below $10^6$/cm$^3$.

Browning and Fontana (*89*) have calculated the theoretical rate of agglomeration of particles in the NS "Savannah" containment system. The authors chose to illustrate their method of calculation by use of the concept of iodine vapor agglomerating to form iodine particles of 0.3-$\mu$ diameter and calculated that 235 sec would be required for this to occur if all the iodine produced in the 69-Mw reactor during 600 days of operation were to be uniformly dispersed in the ship's containment system. Although the authors termed the assumption that iodine agglomerates into particles of pure iodine an artifice, the fact that it is utterly unrealistic to assume that iodine particles large enough to be retained by filters will exist might be overlooked by people unfamiliar with iodine behavior.

It should be recognized that the effect of high specific activity levels, such as may be expected in vaporized fuel particles produced in reactor accidents, on coagulation rates remains to be determined. Some efforts in this direction have been made, but the activity levels were too low to give significant results.

## 4.2 VARIATION OF AIRBORNE IODINE WITH TIME

### 4.2.1 Experiments in Reactor Shells

Deposition experiments in the Zenith, Dido, and Pluto reactor shells have been reported by Megaw and May (*83*) and by Croft and Iles (*90*). Their data showed changing deposition rates during the aging of an iodine aerosol released initially as molecular vapor. The initial airborne iodine concentrations in dry air were 0.01 to 14 $\mu$g/m$^3$ in the Dido and Pluto experiments and 0.3 to 0.8 $\mu$g/m$^3$ in the Zenith shell. The maximum iodine concentrations used were lower than possible accident conditions by a factor of 100 or more and the absence of vaporized fuel particles reduced the degree of accident simulation, but the results obtained are, nevertheless, of considerable interest. Iodine-132 with normal iodine carrier was used in all these experiments. The variation of $^{132}$I concentration with time for six Pluto experiments is shown in Fig. 4.1. Experimental conditions for the Dido and Pluto experiments are given in Table 4.3. Detailed distribution data and Aitken nuclei counts are shown in Fig. 4.2 for

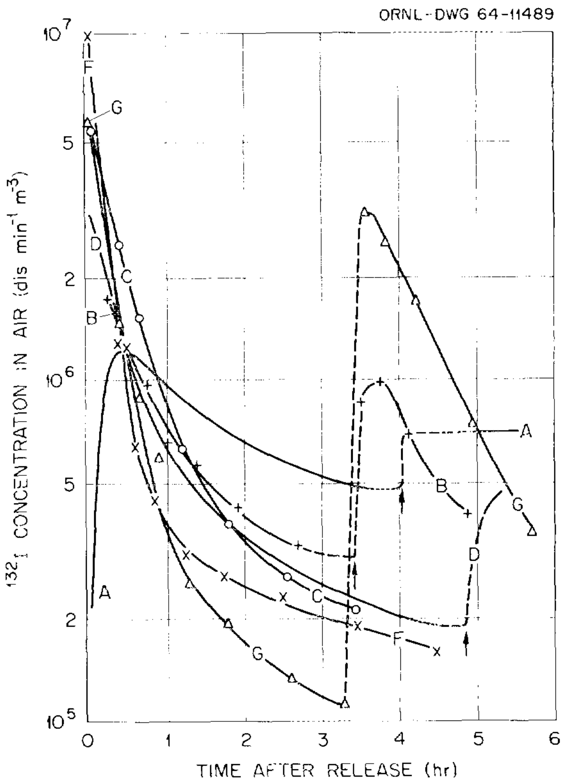


Fig. 4.1. Variation of $^{132}$I Concentration with Time (*83*) (Corrected for Radioactive Decay from $t = 0$).

**Table 4.3. Details of Experimental Releases (83)**

| Reactor | Run | Quantities of Iodine Released | | Calculated Added Concentration | | Maximum Measured Concentration (dpm/m$^3$) at Position 2 | Remarks |
|---|---|---|---|---|---|---|---|
| | | $^{127}$I (mg) | $^{131}$I (mc) | ($\mu$g/m$^3$) | (dpm/m$^3$) | | |
| Dido | 1 | 7 | 26 | 1.0 | $8.2 \times 10^6$ | $4.7 \times 10^6$ | Scrubbers operated |
| | 2 | 1 | 31 | 0.14 | $9.8 \times 10^6$ | $2.8 \times 10^6$ | |
| | 3 | 7 | 31 | 1.0 | $9.8 \times 10^6$ | $2.9 \times 10^6$ | |
| Pluto | A | 0.09 | 28.2 | 0.013 | $8.9 \times 10^6$ | $1.45 \times 10^6$ | |
| | B | 0.95 | 22.9 | 0.14 | $7.2 \times 10^6$ | $1.75 \times 10^6$ | |
| | C | 99 | 30.6 | 14.1 | $9.7 \times 10^6$ | $5.4 \times 10^6$ | |
| | D | 99 | 20.1 | 14.1 | $6.4 \times 10^6$ | $3.0 \times 10^6$ | |
| | E | 7 | 2.7 | 1.0 | $8.5 \times 10^5$ | $9.3 \times 10^5$ | Scrubbers operated |
| | F | 9.9 | 48.9 | 1.4 | $1.5 \times 10^7$ | $1.0 \times 10^7$ | |
| | G | 10 | 39 | 1.4 | $1.2 \times 10^7$ | $5.5 \times 10^6$ | |

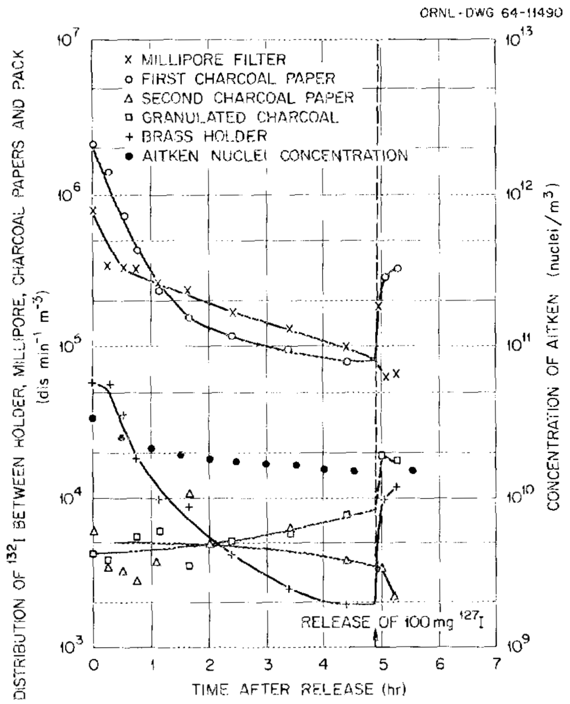


**Fig. 4.2. Run D; Variation of $^{132}$I Concentration with Time (83).**

one experiment (run D) where the initial calculated iodine concentration was 14.1 $\mu$g/m$^3$. As shown in Table 4.3, the maximum measured concentrations were much less than the calculated concentrations (20 to 60%) except in one experiment. This was ascribed to the fact that about 40 min was required for uniform mixing of gases in the reactor shell under the prevailing experimental conditions. Consequently, part of the iodine diffused to the shell wall before mixing was complete.

The sharp breaks in several of the curves shown in Figs. 4.1 and 4.2 are due to liberation of quantities of $^{127}$I vapor in the shell, 10 g in run G, and 100 mg in A, B, and D. The $^{127}$I vapor obviously interchanged with part of the deposited $^{132}$I and caused it to become airborne.

Two charcoal-coated filter papers were used following the Millipore filter in the May Pack in the Pluto experiments; Fig. 4.3 shows the ratio of iodine activity on the first paper to that on the second as a function of time. The ratio decreased rapidly and then became relatively constant at about 20 to 60 after $2\frac{1}{2}$ hr. The authors ascribed this to an unknown vapor compound of iodine, designated "compound X" that has subsequently been identified as organic iodides (see Sect. 3.6). The variation in the fraction of airborne iodine collected on Millipore paper is shown in Fig. 4.4. The data show that this fraction increased

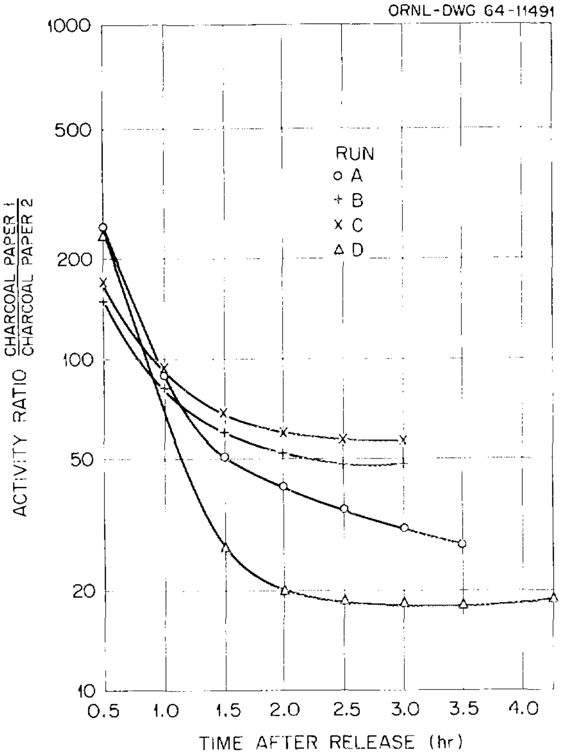


**Fig. 4.3. Variation of the Ratio of Iodine Activity on the First Charcoal-Coated Filter Paper to That on the Second with Time (*83*).**

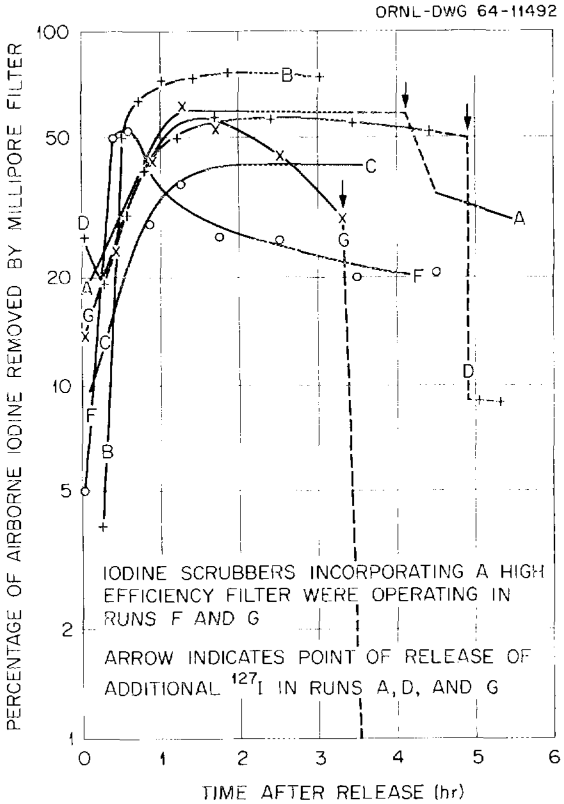


**Fig. 4.4. Percentage of Total Iodine Collected by the Millipore Filter Stages of the Sampler (*83*).**

rapidly during the first hour when molecular iodine was depositing rapidly and then leveled off, or decreased as in runs F and G where the air was being circulated through a high-efficiency filter.

The experimental conditions that prevailed in the Zenith experiments were similar to the Dido-Pluto conditions except for the narrower concentration range (0.3 to 0.8 $\mu g/m^3$), and the same May Pack was used. It is not surprising, therefore, that comparable results were obtained. Data obtained with samples taken in the containment shell are shown in Fig. 4.5. The authors say that in this experiment (No. 2) the first charcoal-impregnated filter paper (ACG/B) and the brass holder together collected any molecular iodine present, with the brass collecting about 20% of the total reaching the sampler. They concluded that from about 4 hr onward the first charcoal paper was sampling about 50% molecular vapor and 50% some other form not sampled by the May Pack and being deposited in the containment shell more slowly than molecular iodine. They also say that the amount of iodine found on the second charcoal paper could be accounted for by a 95% efficiency for the vapor form "compound X."

#### 4.2.2. Experiments in Smaller Enclosures

Croft, Iles, and Davis (*91*) describe experiments to determine the deposition characteristics of airborne iodine released initially as molecular vapor in quantities appropriate for power-reactor operations if all the iodine generated during long-term operation is assumed to be released. The release was made into a room about 30 $m^3$ in volume. They measured the deposition characteristics from an aging aerosol in both saturated and unsaturated air at ambient temperatures and with dry and wet surfaces. Figure 4.6 shows the

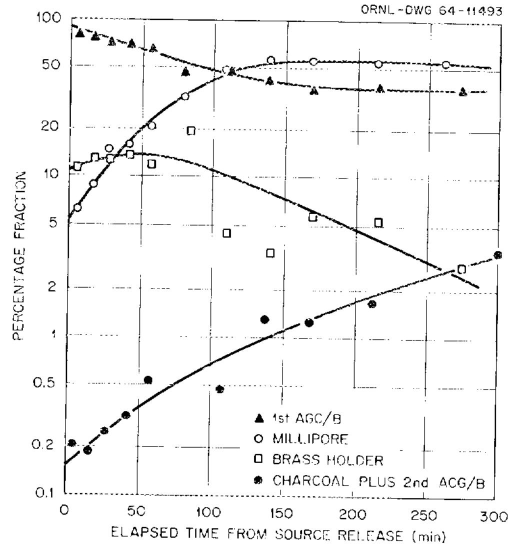


**Fig. 4.5. Distribution of Airborne Iodine Expressed as Percentage of the Total (*90*).**

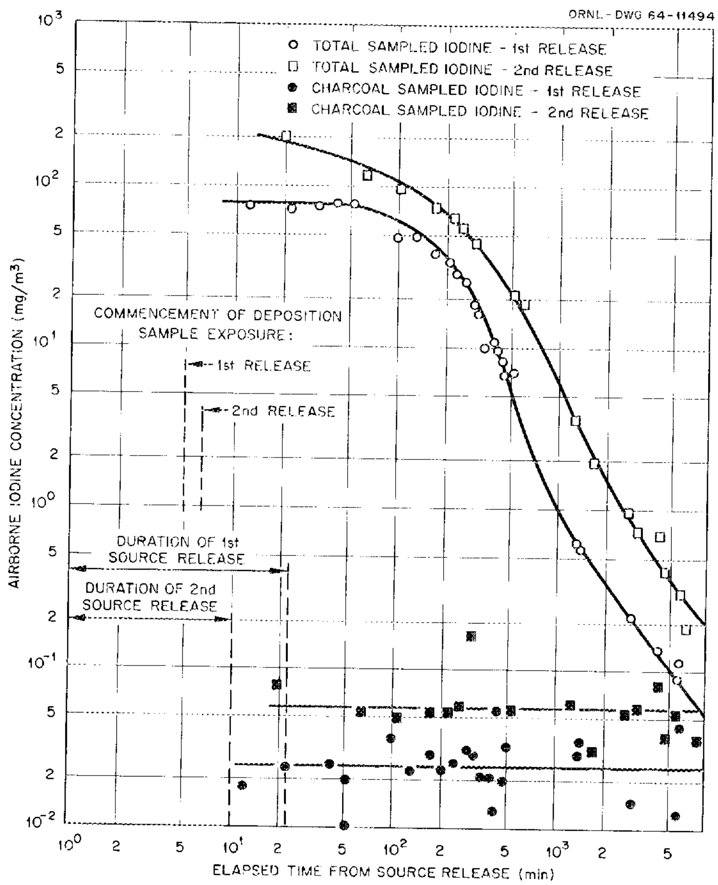


**Fig. 4.6. Total and Charcoal-Sampled Airborne Iodine Concentrations in the Two Dry Experiments (*91*).**

variation of airborne iodine with time during the dry experiments, while Fig. 4.7 gives results of the wet experiment. The general pattern is seen to be similar in both cases. The airborne iodine concentration, after initial dispersion, falls steadily by deposition to ~1% of the initial value after 1 day, leakage from the containment being negligibly small during this period.

A log/linear plot, as in Fig. 4.8, shows that in each of the three releases the time dependence

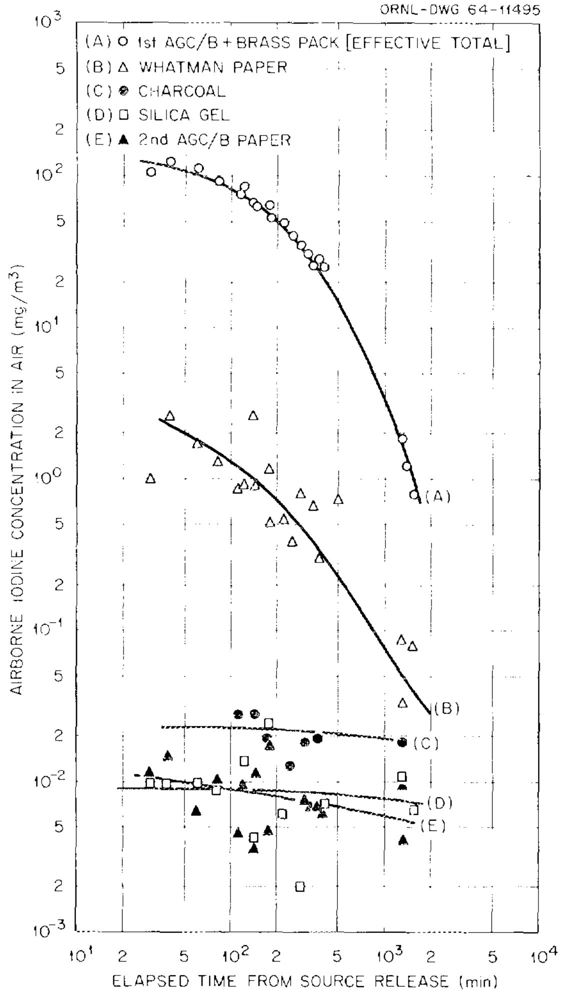


**Fig. 4.7. Airborne Iodine Concentrations as Sampled by the Separate Components of the Sampling Pack in the Wet Experiment (*91*).**

of iodine concentration can adequately be described by a negative exponential function over an initial period of approximately 8 to 10 hr, whereafter the rate of attenuation decreases. Evaluation of the three exponential curves shown in Fig. 4.8 is given in Table 4.4.

The measured deposition velocities are very close to those at related stages of the Zenith, Dido, and Pluto containments when the airborne iodine concentrations were several orders of magnitude (~$10^5$) less. This suggests that the main factor limiting sorption is gas-phase diffusion to the surface.

Parker *et al.* (*46*) give data, shown in Table 4.5, on the variation of airborne iodine in a small (6.7-$ft^3$) stainless steel tank. The aerosol was produced by melting in air a stainless steel $UO_2$ fuel specimen that had been irradiated to 7000 Mwd/ton. The data show that the iodine concentration in the tank dropped rapidly during the first few minutes after melting, as might be expected for molecular iodine in a vessel having a large surface-to-volume ratio. The concentration then remained relatively constant for almost 2 hr; then a large fraction of the iodine that had deposited on the walls of the container was removed by passage of 1.5 tank volumes of argon through it.

Croft *et al.* (*91*) reported data (Table 4.6) showing that the effective deposition velocity for iodine on different surfaces varied with the time interval over which the measurements were made.

The 0–20-min values of effective deposition velocity in Table 4.6 are those most nearly applicable to molecular iodine vapor, in which form a major part of the source was deposited in the early stages. The decreased deposition velocity during the first 500 min indicates the influence of forms other than molecular iodine after the initial 20-min period. The data indicate that equilibrium was very nearly established during the first 500 min, with little change in rate occurring during the final 1000-min exposure period.

Data on the deposition of iodine, released as molecular iodine, in an air-filled stainless steel tank (1350 $ft^3$) are shown in Fig. 4.9 (*47*). The air in the tank was not stirred during this test, and it was unfiltered, with normal humidity. Two different deposition rates are distinguishable; the slower rate seemed to be constant over a rather extended period. The first curve extrapolates to an initial value of 1.25 mg/m$^3$, about half the value calculated assuming 100% transfer of iodine from the furnace to the containment tank and uniform dispersion in the tank.

Results of a simpler experiment (*92*) in a smaller (6.7-ft$^3$) stainless steel tank are shown graphically in Fig. 4.10. Molecular iodine in this case was released into the tank filled with a steam-air mixture at 30 psig total pressure. The half-life of deposition varied from an initial value of about 6 min to 2100 min over the last 240 min of the exposure period.

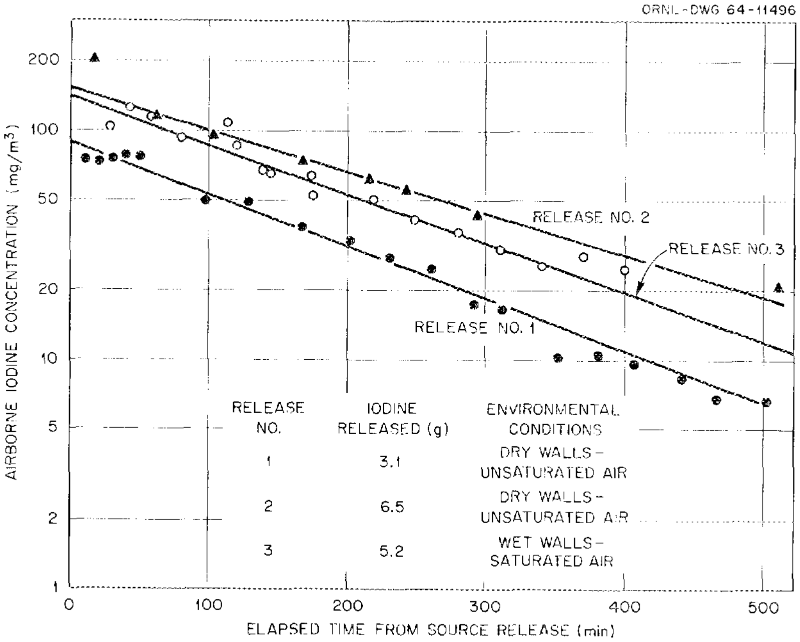


**Fig. 4.8. Total Sampled Airborne Iodine Concentration for the First 8 hr Following Each Release (*91*).**

**Table 4.4. Evaluation of the Exponential Attenuation of Airborne Iodine Concentration (*91*)**

| Experiment Number | Environmental Conditions | Mass Released (g) | Mass Released / Volume of Containment (mg/m$^3$) | Exponential Function for Concentration, C (mg/m$^3$), Against Time, *t* (min) | Effective Deposition Velocity,[a] *V* (cm/sec) |
|---|---|---|---|---|---|
| 1 | Dry | 3.1 | 113 | $C = 90e^{-0.693t/130}$ | $4.0 \times 10^{-3}$ |
| 2 | Dry | 6.5 | 237 | $C = 150e^{-0.693t/170}$ | $3.1 \times 10^{-3}$ |
| 3 | Wet | 5.2 | 190 | $C = 140e^{-0.693t/140}$ | $3.7 \times 10^{-3}$ |

[a]The effective deposition velocity, *V*, is derived from the experimental exponential decay constant, $\lambda$, by the relation $\lambda = vA/V$, where *A* is the effective deposition surface area and *V* is the free volume of the containment.

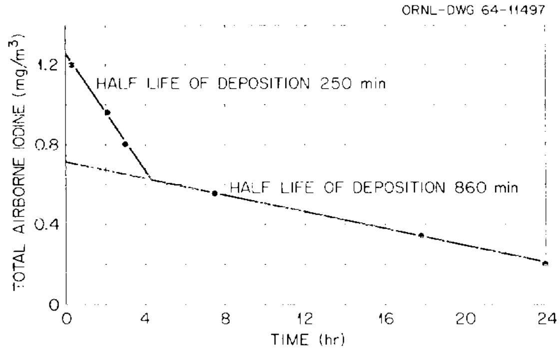


Fig. 4.9. Deposition of Iodine in the Stainless Steel Containment Vessel of the NSPP – Molecular Iodine Released in Unstirred Air.

Table 4.5. Variation of Airborne Iodine Released by Melting Irradiated (7000-Mwd/ton) Stainless Steel $UO_2$ (*46*)

| Time After Melting (min) | Percent of Iodine Inventory Airborne |
|---|---|
| 0 | 91[a] |
| 10 | 11 |
| 25 | 21 |
| 55 | 5 |
| 115 | 6 |
| 118 | 11 |
| 300 | 72[b] |

[a]Calculated from total amount released to the confinement vessel.

[b]Calculated from total amount collected in the filter-absorber system.

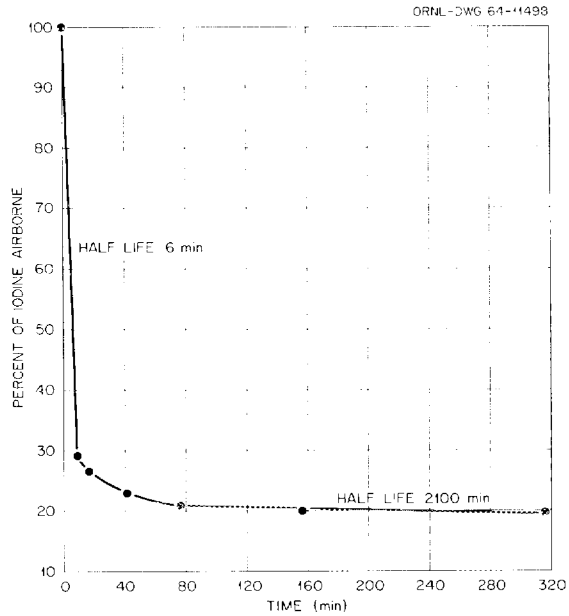


Fig. 4.10. Deposition of Iodine in Steam-Air Filled Stainless Steel Tank (6.7 ft³).

Table 4.6. Values of the Effective Deposition Velocity for Varying Exposure Times (*91*)

| Time from Source Release (min) | Exposure ($x$) (mg min$^{-1}$ m$^{-3}$) | Effective Deposition Velocity from $t = 0$ for Release No. 1 ($y/x$) (cm/sec) | | | |
|---|---|---|---|---|---|
| | | Mild Steel | Concrete | Paint[a] | PTFE[b] |
| | $\times 10^3$ | $\times 10^{-1}$ | $\times 10^{-2}$ | $\times 10^{-3}$ | $\times 10^{-3}$ |
| 0–20 | 1.5 | 6.2 | 6.7 | 7.2 | 5.0 |
| 0–100 | 7.7 | 2.0 | 6.7 | 3.7 | 1.1 |
| 0–500 | 17 | 1.13 | 6.7 | 2.7 | 0.68 |
| 0–1500 | 19 | 1.06 | 6.1 | 2.6 | 0.64 |

[a]Chlorinated-rubber-based paint on a hard asbestos surface.

[b]Polytetrafluoroethylene.

## 4.3 PARAMETRIC EFFECTS ON DEPOSITION OF IODINE ON SURFACES

A great deal of attention has been devoted to study of factors affecting the rate of deposition of iodine on surfaces. The chemical form of iodine used for the study obviously will determine, in part, deposition rates. However, atmospheric and surface conditions also affect this rate, and these factors are discussed in this section. Interaction of variables undoubtedly occurs, and the contribution of such interactions to observed iodine behavior cannot be readily evaluated in some cases. It should also be recognized that while this section is devoted primarily to deposition studies, desorption occurs simultaneously with sorption except for surfaces that are a perfect sink for all forms of iodine present during the experiment. Although some overlap results from treatment of desorption in a separate section, this procedure can be justified on the basis that conditions existing during many desorption measurements were different from those under which deposition occurred.

### 4.3.1 Surface Composition

Croft *et al.* (*91*) conducted experiments on deposition of high concentrations of iodine (100–200 mg/m$^3$) on various surfaces. Deposition was found to be dependent on the nature of the surface, with bare mild steel and concrete surfaces showing higher deposition values than painted surfaces, which in turn were more receptive than the plastic (PTFE and polyethylene) surfaces (Table 4.7). Oxidized or polished mild steel surfaces were similar to each other in regard to deposition behavior. The metal surfaces were rapidly coated with a reddish corrosion layer. They were able to load surfaces to the high iodine levels indicated below (in mg/m$^2$) without apparent saturation: mild steel, 2000; concrete, 1400; paint, 75; PTFE, 17. However, there was some evidence of a reduced deposition rate at the higher loading levels.

No significant differences were noted in deposition on five sets of dry painted surfaces. This is in direct contrast to the measurements made under wet conditions, which showed faster deposition

Table 4.7. Effective Deposition Velocity on Surfaces (*91*)

| Surface | Effective Deposition Velocity for 1500 min (cm/sec) | |
|---|---|---|
| | Dry | Wet |
| Room surfaces | $4.0 \times 10^{-3}$ | $3.7 \times 10^{-3}$ |
| Mild steel | $10.6 \times 10^{-2}$ | $5.8 \times 10^{-2}$ |
| Concrete | $6.1 \times 10^{-2}$ | $4.0 \times 10^{-2}$ (est.) |
| Mean of all painted surfaces | $2.6 \times 10^{-3}$ | |
| Mild steel with chlorinated-rubber-based paint | | $8.0 \times 10^{-3}$ |
| Concrete with chlorinated-rubber-based paint | | $3.3 \times 10^{-3}$ |
| Hard asbestos with chlorinated-rubber-based paint | | $5.6 \times 10^{-3}$ |
| Mild steel with epoxy-resin-based paint | | $1.6 \times 10^{-2}$ |
| Concretes with epoxy-resin-based paint | | $1.7 \times 10^{-2}$ |
| Polytetrafluoroethylene (PTFE) | $6.4 \times 10^{-4}$ | $1.9 \times 10^{-3}$ |
| Polyethylene sheet | | $7 \times 10^{-3}$ |
| 480 cm$^3$ of water static in a vitreous enameled experimental dish, surface area 680 cm$^2$ | | $2.5 \times 10^{-3}$ |

Table 4.8. Sorption of Iodine in Presence of Air at Room Temperature (*93*)

| Material | Expt. No. | $I_2$ Conc. in Air ($\mu g/m^3$) | Rate Coefficient of Sorption in Linear Part of Curve (cm/sec) | Amount of Sorbed Iodine at Plateau of Curve ($\mu g/cm^2$) |
|---|---|---|---|---|
| Concrete | 4 | 20 | 0.1 | |
| | 5 | 2500 | 0.08 | 70(?) or higher |
| Granite | 4 | 20 | 0.020 | |
| | 5 | 2500 | 0.0032 | 2 |
| Acrylate | 4 | 20 | 0.012 | |
| | 5 | 2500 | 0.003 | 1.8 |
| Polyvinyl chloride | 4 | 20 | 0.005 | |
| | 5 | 2500 | 0.0004 | 0.2 |

rates onto epoxy-resin-based paint. This will be discussed further in the section on effects of moisture.

Forberg *et al.* (*93*) exposed small samples of granite, plastic materials, and concrete to elemental iodine vapor. Their results are reported in Table 4.8 (a rearrangement of the published table). The rate of sorption on concrete was independent of iodine loading up to 50 $\mu g/cm^2$. Forberg's results show relationships of the various materials which are similar to those obtained by Croft *et al.* (*91*), Table 4.9, third column, and by Hudswell *et al.* (*94*).

The results of a number of investigations are summarized in Table 4.10 (*81*). The results are arranged in the order of increasing dosage of iodine. The period of exposure of the absorbing surfaces to iodine was 60 min, except in the experiments referred to in columns 8 and 9 of Table 4.10, where it was approximately 200 min.

The amount of iodine sorbed on the copper samples varied by nearly a factor of $10^6$ in these experiments, but the rate coefficient of uptake (velocity of deposition) varied by less than a factor of 10.

In the lower part of Table 4.10, the amount of iodine sorbed in the various surfaces is expressed as a percentage of that sorbed on copper.

The sorption on silver was somewhat higher than that on copper, except in the experiments of columns 6 and 7, where it was rather less.

Table 4.9. Velocity of Deposition of Iodine on Surfaces in Containment Vessels (*81*)

| Surface | Velocity of Deposition (cm/sec) | |
|---|---|---|
| | Pluto[a] | Winfrith[b] |
| Copper | 0.30 | |
| Charcoal paper | 0.28 | |
| Mild steel | 0.24 | 0.22 |
| Painted steel | 0.18 | 0.0039 |
| Aluminum | 0.14 | |
| Concrete | | 0.055 |
| Polyethylene | 0.028 | |
| PTFE | | 0.0012 |

[a] 0.35 to 0.36 $\mu g$ of $I_2$ per $m^3$ of air.
[b] $7.8 \times 10^4$ to $1.6 \times 10^5$ $\mu g$ of $I_2$ per $m^3$ of air.

The sorption on mild steel was nearly as high as that on silver and copper in the Pluto experiments, and this was also found in wind tunnel experiments in which the iodine carrier level was low.

At higher iodine concentrations and especially at higher temperatures and in dry air, the sorption on mild steel was only a few percent of that on silver and copper.

**Table 4.10. Relative Deposition of Iodine on Various Surfaces (*81*)**

| | Pluto ($7000\ m^3$) Col 1 | Col 2 | Laboratory Apparatus | | Col 5 | Laboratory Apparatus | | Laboratory Apparatus | | Winfrith 27-$m^3$ Room Col 10 |
|---|---|---|---|---|---|---|---|---|---|---|
| | | | Col 3 | Col 4 | | Col 6 | Col 7 | Col 8 | Col 9 | |
| Period of exposure, min | 60 | 60 | 60 | 60 | 60 | 60 | 60 | ~200 | ~200 | 60 |
| $I_2$ vapor concentration, $\mu g/m^3$ | 0.35 | 40–100 | 100–1000 | $10^3$–$10^4$ | $10^4$–$10^5$ | 700 | $4 \times 10^4$ | $2 \times 10^4$ | $4 \times 10^4$ | 8–16 $\times 10^4$ |
| Atmosphere | Moist air | Dry air | Dry air | Dry air | Dry air | Dry air | Dry air | Steam + air | Slightly moist air | Moist air |
| Temperature, °C | Ambient | Ambient | Ambient | Ambient | Ambient | 200 | 200 | 150 | 150 | Ambient |
| Absorbing surfaces | 2-in. disks | ⅛-in. rods | ⅛-in. rods | ⅛-in. rods | ⅛-in. rods | ⅛-in. rods | ⅛-in. rods | 2-in. disks | 2-in. disks | 2¼-in. disks |
| $I_2$ sorbed on copper, $\mu g/cm^2$ | $3.6 \times 10^{-4}$ | $\sim 10^{-1}$ | ~1 | ~10 | ~30 | 4 | 240 | 83 | 150 | 80[a] |
| Velocity of deposition on copper, cm/sec | 0.29 | 0.6 | 0.6 | 0.4 | 0.2 | 1.2 | 1.5 | 0.3 | 0.3 | 0.22[a] |
| Amounts sorbed relative to copper = 100 | | | | | | | | | | |
| Copper | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | 100 | |
| Silver | | 128 | 102 | 105 | 173 | 74 | 88 | 150 | 144 | |
| Charcoal paper | 110 | | | | | | | | | |
| Mild steel | 83 | 28 | 56 | 32 | 14 | 13 | 2 | 39 | 0.4 | 100[b] |
| Stainless steel | | | | | | | | 18 | 1.7 | |
| Painted steel | 52 | | | | | | | 4.5 | 4.0 | 1.8[b] |
| Concrete | | | | | | | | | | 25[b] |
| Aluminum | 62 | 3.0 | 2.8 | 3.4 | 42 | 3.6 | 0.35 | 1.7 | 0.15 | |
| Magnesium | | 1.9 | 4.4 | 3.4 | 6.9 | 1.7 | 0.30 | | | |
| Graphite | | 3.6 | 8.2 | 4.2 | 3.1 | 1.4 | 0.36 | | | |
| Glass | | 0.13 | 0.25 | 0.78 | 0.87 | 1.2 | 0.06 | 1.8 | Nil | |
| Polythene | 9 | | | | | | | | | |
| PTFE | | | | | | | | | | 0.5[b] |

[a]Amount sorbed on mild steel.

[b]Relative to mild steel = 100.

Table 4.11. Iodine Sorption on Steel Surfaces in Filtered Air at 15 psig and Ambient Temperature (*46*)

| Exposure Time (min) | Iodine Sorbed ($\mu\mu g/cm^2$) | | |
|---|---|---|---|
| | Stainless | Mild[a] | Painted |
| 6 | 3 | 20 | 9 |
| 14 | 4 | 114 | 58 |
| 34 | 11 | 286 | 174 |
| 74 | 11 | 529 | 267 |
| 134 | 19 | 550 | 413 |
| 214 | 35 | 678 | 535 |

[a]Preoxidized by exposure to steam at a high temperature to produce a black oxide film on the surface typical of that found in high-temperature water systems.

The sorption on aluminum, magnesium, and graphite surfaces at ambient temperatures was also found to be generally a few percent of that on copper. The rate coefficient for sorption on these materials was of the order of 0.001 to 0.005 cm/sec.

A monolayer coverage of iodine is about 0.3 $\mu g/cm^2$. Copper, silver, charcoal paper, and concrete apparently sorb many monolayers of iodine with little reduction in the rate of deposition. Mild steel at ambient temperature and in moist atmospheres may belong in this group.

Stainless steel, aluminum, magnesium, and graphite adsorb about a monolayer of iodine. Glass, painted surfaces, and plastics adsorb from one to ten monolayers.

The rate of sorption of iodine from air at ambient temperature on different types of steel surfaces was determined as a function of time, and the results are recorded in Table 4.11 (*46*). The uptake of iodine increased over the time period studied. In contrast, similar specimens exposed in a steam-air atmosphere at initial temperatures ranging from 100 to 120°C showed comparatively little change in iodine sorption with exposure times up to 5 hr, and the amounts sorbed on all three types of surfaces were much less than in dry air.

### 4.3.2 Oxidation Conditions

A comparison of the location of the fission-product iodine released from uranium-aluminum alloys in three atmospheric environments, air, helium, and an 80% steam–20% air mixture, is shown in Table 4.12 (*84*). This type of study is of value as a measure of the transport efficiency and, in the case of the steam atmosphere, of the washout effect resulting from condensation of steam. A comparison of the apparent chemical effect of each atmosphere is also of interest. The data show that up to 88% of the iodine released in air was apparently in the molecular form and was carried through the filters to charcoal absorbers. In helium, on the other hand, the percentage of iodine reaching the charcoal was 0.1% or less with the bulk having deposited in the furnace tube and on the absolute filters, indicating that almost all the iodine was in the reduced form.

Davies *et al.* (*95*) found that stainless steel retained iodine at temperatures up to 300 to 400°C when helium or nitrogen was used as the carrier gas (see also Sect. 2 of this report).

Castleman (*96*) studied the deposition of iodine as a function of surface temperature (discussed in the next section) in different atmospheres. He found that most of the iodine deposited on hot (250–350°C) surfaces in helium but in air less than 3% deposited at temperatures above 100°C. Additional experiments by Castleman (*28*) concerning the chemical and physical behavior of the released fission products iodine, cerium, barium, lanthanum, molybdenum, and tellurium released from metallic uranium in purified helium and in air are described in Sect. 3.4.2 of Chap. 3, "Radioactivity Generation, Release, and Transport," of the *Reactor Containment Handbook*.

Raines *et al.* (*97*) reported results of experimental and theoretical studies of fission-product deposition from helium circulating in a closed stainless steel loop. They state that correlation of iodine deposition data was not attempted because of their complex nature.

Castleman and Salzano (*98*) reported that iodine, released from uranium metal in a helium atmosphere, deposits on gold, stainless steel, and quartz surfaces at 250 to 350°C in a combined form, probably as a uranium compound. The comparative performance of paints may depend on humidity, as well as on surface temperature as discussed in Sect. 4.3.5.

Parker *et al.* (*92*) have compared the sorption of iodine on painted steel at about 120°C with that on preoxidized mild steel and stainless steel

Table 4.12. Distribution of Fission-Product Iodine Released from Uranium-Aluminum Alloys in Various Atmospheres (*84*)

| Transport Fraction | Fuel Temp. (°C) | Percent of Total Inventory | | |
|---|---|---|---|---|
| | | Air | Helium | Steam-Air |
| Total release | 700 | 37.8 | | 27.0 |
| | 800 | 78.6 | 29.8 | 76.8 |
| | 900 | 91.9 | 52.8 | 90.6 |
| | 1000 | 97.3 | 82.1 | 95.6 |
| | 1100 | 98.4 | 82.4 | 96.8 |
| | 1150 | 94.2 | | |
| In hot zone (iodide form) | 700 | 19.9 | | 17.0 |
| | 800 | 30.6 | 28.4 | 9.3 |
| | 900 | 11.3 | 51.3 | 12.9 |
| | 1000 | 38.2 | 69.6 | 13.0 |
| | 1100 | 6.3 | 65.8 | 10.3 |
| | 1150 | 17.4 | | |
| Transported to filters by particles | 700 | 2.8 | | 0.3 |
| | 800 | 4.5 | 1.4 | 0.1 |
| | 900 | 4.2 | 1.5 | 0.7 |
| | 1000 | 0.9 | 13.3 | 4.1 |
| | 1100 | 3.9 | 17.4 | 3.9 |
| | 1150 | 9.2 | | |
| Absorbed in charcoal traps (principally molecular iodine) | 700 | 15.0 | | 0.002 |
| | 800 | 44.2 | 0.08 | 0.05 |
| | 900 | 76.4 | 0.03 | 0.7 |
| | 1000 | 59.0 | 0.11 | 8.5 |
| | 1100 | 88.0 | 0.06 | 9.6 |
| In the steam condensate | 700 | | | 9.8 |
| | 800 | | | 66.9 |
| | 900 | | | 77.0 |
| | 1000 | | | 69.8 |
| | 1100 | | | 73.0 |

at 200 and 300°C in a steam-air atmosphere. No difference was found in the amount sorbed in the stainless steel surfaces at the two temperatures (3 $\mu\mu g/cm^2$) but the sorption on mild steel was slightly less at the higher temperature (22 $\mu\mu g/cm^2$) and both had less iodine than the painted surface (90 $\mu\mu g/cm^2$).

#### 4.3.3 Temperature

Croft *et al.* (*91*) found that concrete retained a high proportion of iodine at temperatures to 300°C and only lost 12% during a 24-hr test at 200°C. Davies *et al.* (*95*) found that carrier-free iodine was retained on stainless steel at temperatures of 300–400°C when helium or nitrogen was used as a carrier gas (see also Sect. 2). The absence of oxygen appears to be important in this case. Copper and silver adsorbed more iodine at higher temperatures than at room temperature as shown by the data (*81*) in Table 4.13 (a rearrangement of the published table). This effect was also noted by Browning (*99*). At higher iodine concentration levels and especially at high temperatures in dry air, the sorption on mild steel was only a few percent of that on silver or copper.

Table 4.13. Comparison of Sorption of Iodine on Various Surfaces in Air at 20°C and 200°C (Period of Sorption, 1 hr) (*81*)

| $I_2$ Conc. ($mg/m^3$) | Temp. (°C) | $I_2$ on Surface ($\mu g/cm^2$) | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | Glass | Mg | Graphite | Al | Mild Steel | Cu | Ag |
| 0.50 | 20 | 0.015 | 0.031 | 0.075 | 0.037 | 0.50 | 0.90 | 0.81 |
| 0.73 | 200 | 0.043 | 0.21 | 0.063 | 0.107 | 0.36 | 2.0 | 1.6 |
| 37.2 | 20 | 0.40 | 1.96 | 1.32 | 1.15 | 1.52 | 17.2 | 40 |
| 46.5 | 200 | 0.45 | 1.83 | 1.00 | 1.32 | 5.7 | 108 | 90 |

### 4.3.4 Concentration

Croft *et al.* (*91*) state that in their experiments with high concentrations of iodine (100–200 $mg/m^3$) the measured deposition velocities were very close to those observed at related stages in the Zenith, Dido, and Pluto experiments, where the airborne iodine concentrations were several orders of magnitude (approximately $10^5$) less. This suggests that the main rate-determining factor in sorption is the rate of gas-phase diffusion to the surface.

The rate of sorption on painted steel surfaces appears to be influenced by the amount of iodine available. The adsorption of copper, silver, charcoal paper, and concrete proceeds with little fall in rate when hundreds of monolayers of iodine have been adsorbed. This is also true for adsorption on mild steel in the presence of moisture. Some high-surface-area systems lose adsorption efficiency at higher iodine loadings in dry air. Glass, painted surfaces, and plastics are limited to a few monolayers of iodine.

### 4.3.5 Moisture

An accident with a water-cooled and/or -moderated reactor leading to a major release of volatile fission products would result in the simultaneous release of large quantities of steam into the secondary containment and in heavy condensation on containment surfaces. The results of some small-scale experiments by Morris and Nicholls (*100*) on the sorption of iodine on various materials at 150°C in dry air and in an air-steam mixture are summarized in terms of the rate coefficient (deposition velocity) in Table 4.14. There was a much higher deposition velocity of iodine on mild steel in the presence of steam. Thus, unpainted mild steel is an effective getter for molecular iodine in the 150°C temperature range with steam atmospheres (no water condensation).

Table 4.14. Uptake of Iodine by Surfaces During 4 hr at 150°C (*100*)

| Material | Deposition Velocity (cm/sec) | |
|---|---|---|
| | 40% Steam | Dry |
| Copper | 0.31 | 0.30 |
| Mild steel | 0.12 | 0.0013 |
| Silver | 0.47 | 0.43 |
| Aluminum | 0.0060 | 0.00046 |
| Stainless steel | 0.054 | 0.0050 |
| Painted mild steel | 0.014 | 0.012 |
| Glass | 0.0055 | 0.00001 |

Parker *et al.* (*84*) reported that steam washout in their experiments accounted for up to 75% of the iodine; even with 20 vol % air present in the saturated steam, only 10% of the iodine remained volatile enough to reach the charcoal filter system. These workers also compared the distribution of fission-product iodine released from irradiated aluminum-uranium alloys in steam and dry atmospheres (*85*), and the data are given in Table 4.15.

Table 4.15. Distribution of Iodine Released from Molten Aluminum-Uranium Alloy in Moist and Dry (Ordinary Humidity) Air (*85*)

| Location of Activity | Percent of Released Iodine in Each Location | |
|---|---|---|
| | Dry Air | Moist Air |
| Furnace tube | 1.8 | 3.9 |
| Aerosol tank | 82.0 | 48.1 |
| Steam condensate (30 ml) | | 6.7 |
| Transite pipe (12 in. long) | 1.2 | 3.1 |
| Filters | 6.9 | 8.9 |
| Charcoal beds | 8.0 | 29.2 |

It was indicated in Sect. 4.3.1 that there were no systematic differences among the various painted surfaces with dry conditions. Under wet conditions a measurably faster deposition rate onto epoxy-resin-based paints has been observed. This behavior is consistent with measurements by Forberg *et al.* (*93*) on epoxy and chlorinated rubber.

### 4.3.6 Carrier Gas Velocity

The rates at which carrier gas flows past fuel samples during release of iodine is generally considered to have no effect on the rate of release. Browning *et al.* (*40*) reported data showing that varying the rate at which helium flowed past fuel samples in in-pile experiments from 60 to 350 fpm did not affect the release of iodine from $UO_2$ specimens or from the high-temperature zone. However, flow conditions, whether laminar or turbulent, do affect the deposition patterns of iodine in off-gas piping or duct systems. Laminar conditions are applicable to the determination of the forms of iodine with diffusion tubes, as previously described in Sect. 3. Higher deposition rates were reported by Raines *et al.* to occur at bends and rough surfaces in loop equipment as a result of increased gas turbulence (*101*). Behavior of iodine on particulates will be influenced by the flow patterns in a reactor system.

Results of experiments (*102*), showing distribution of iodine released into a stainless steel vessel in the form of molecular iodine, are summarized in Table 4.16. Five release experiments (runs) were conducted: filtered air; filtered air and steam with two containment times; and filtered air and steam with two types of organic gases added. In all five runs, the tank contained filtered air at 15 psig. In all runs except 6-11, the tank also contained 10 to 15 psi of steam. In four runs, pressure in the tank was released after 5 hr; in run 8-4 the pressure was held for 18 hr. The results are tabulated below. In clean air a fairly high percentage (80%) of the iodine rapidly plated out on the tank walls. Only 3.0% remained available then for leakage from the pressurized tank. In this case, however, considerable desorption occurred later, bringing the total airborne quantity to 13%. This amount decreased in the presence of steam to 3–6%, while about 50% of the iodine was collected in the condensate formed as the tank cooled over a period of several hours. On the other hand, when organic materials were added, as in run 7-16 where dry ice ($CO_2$) and acetone were obviously present in the sample and in run 9-10 where a 500-$cm^3$ sample of freshly prepared mixed hydrocarbons (from the hydrolysis of $UC$-$UC_2$) was added to the tank, a decided increase (to 27%) in the amount remaining airborne occurred. The data in Table 4.16 show that factors other than the presence of moisture have a marked effect on the amount of iodine available for leakage from the simulated containment shell after aging of the aerosol.

An attempt to classify the forms of iodine in each test by means of an elaborate filter pack showed no significant trends, except that in each of the two organic runs a large fraction (16%) of the iodine penetrated both filters and the metallic silver or copper gauze, suggesting nonreactive iodine.

Unfortunately, other supporting data are not sufficiently clear to permit unequivocal identification of the nonreactive iodine. The amount of iodine in a form sufficiently nonreactive to penetrate 1.5 in. of charcoal remained low in all cases, although it was noticeably higher in one experiment in the presence of organic gas.

High-velocity deposition distributions are being interpreted in terms of diffusion, interception, and inertial impaction of aerosols in fibrous filters by Browning and Silverman (*103*). See Sect. 5, "Trapping of Fission-Product Iodine and Induced Removal Processes."

Table 4.16. Iodine Deposition and Desorption in the Containment Mockup Facility (Expressed as Percentage of Total Iodine Released into the Tank)

| | Air | Steam-Air | | Steam-Air and Organics | |
|---|---|---|---|---|---|
| | Run 6-11 | Run 6-25[a] | Run 8-4 | Run 7-16 | Run 9-10 |
| Iodine held in containment tank | | | | | |
| Retained on tank walls | 79.6 | 60.8 | 38.2 | 19.3 | 20.9 |
| Collected in steam condensate | | 34.9 | 54.0 | 52.9 | 47.3 |
| Total retention | 79.6 | 95.7 | 92.2 | 72.2 | 68.2 |
| Iodine removed from tank | | | | | |
| By pressure release | 3.0 | 1.1 | 2.4 | 12.9 | 8.5 |
| By argon displacement | 7.4 | 1.2 | 2.4 | 13.3 | 15.9 |
| By air sweep | 2.9 | 0.3 | 1.7 | 0.5 | 2.6 |
| Total removed, airborne | 13.3 | 2.6 | 6.5 | 26.7 | 27.0 |
| Iodine removed on test samples | 6.1 | 1.7 | 1.4 | 0.8 | 2.2 |
| Distribution of airborne iodine from tank | | | | | |
| Retained on filters | 4.1[b] | 0.5 | 1.5 | 0.1 | 0.3 |
| Retained on silver/copper screens | 5.3 | 0.8 | 2.3 | 6.4 | 8.5 |
| Retained on charcoal papers | 3.0 | 0.4 | 0.4 | 2.0 | 15.8 |
| Retained in charcoal cartridges | 0.3 | 0.6 | 2.0 | 16.4 | 2.3 |
| Loss through 1.5 in. of charcoal | 0.003 | 0.0002 | 0.05 | 0.3 | 0.03 |

[a] In 6-25, iodine was released in steam and air under conditions providing a large amount of condensate (uninsulated tank).

[b] One organic membrane filter used in this test probably reacted with and retained some molecular iodine in addition to particulate iodine.

### 4.3.7 Surface-to-Volume Ratio

When comparing the results of experiments in large reactor vessels with data obtained in smaller-scale laboratory experiments, the surface-to-volume ratios should be considered in the interpretation of the physical and chemical behavior of iodine, since the half-life of iodine in the gas phase will depend to some extent on this ratio. Literature data are sometimes difficult to correlate in this respect because unstated or unmeasured amounts of surfaces other than that of the shell itself were present during the experiments. Ratios calculated for several present or proposed experimental facilities (given in Table 3.1) can be compared with the value of 0.033 $ft^{-1}$ calculated for a typical large reactor shell (Dresden).

A velocity of 0.1 to 0.3 cm/sec for molecular iodine depositing on the metallic and concrete surfaces of a containment vessel has been indicated by Chamberlain (*81*). In the Dido and Pluto reactor containment shells, the surface-to-volume ratios were approximately 0.004 $cm^{-1}$. The initial velocity of deposition was 0.15 cm, and the initial half-life deposition on the surface of the reactor containment was $0.693/(0.004 \times 0.15) = 1000$ sec. The effective deposition velocity ($V$) in containment vessels can be derived from the experimental exponential decay constant by the relation $\lambda = VA/V$, where $A$ is the effective surface area and $V$ is the free volume of the containment.

## 4.4 DESORPTION

It was pointed out earlier in this section that desorption in many cases is occurring to some extent while iodine is depositing. It is important to know what fraction of the iodine that has deposited on a surface may become airborne again and the conditions affecting this fraction. The factors affecting desorption will be discussed here.

#### 4.4.1 Oxidation of Iodides

Data presented in Sect. 2 show that iodides have much lower vapor pressure than molecular iodine. Consequently, when iodine deposited on surfaces in the reduced state is exposed to atmospheric oxygen, partial desorption frequently occurs. Castleman (*96*) reported that when surfaces on which iodine from irradiated uranium or U-Mo alloys deposited from a helium stream at 250 to 350°C were exposed to air at the same temperature, the deposited iodine was slowly converted to the elemental form. He stated that the iodine probably deposited as $UI_2$ since CsI is stable in air at these temperatures. In reactor accidents involving pressurized water reactors, reducing conditions may exist in the early phases of the accident due to metal-water reactions. Later, as steam condensation in the shell results in lower pressure, air may enter the shell. The ease with which the reduced iodine will oxidize depends on a number of factors, but one of the more important is the element with which it is combined. There is very little published information on this subject.

#### 4.4.2 Surface Effects and Exchange Reactions

Croft *et al.* (*91*) determined rates of iodine desorption from materials exposed to high concentrations of molecular iodine (see Sect. 4.4.3 for experimental details) and reported the following values for the half-life of desorption in air at ambient temperatures: mild steel, 17 days; concrete, 10 days; painted steel, 4 days; polytetrafluoroethylene (PTFE), 2 days. A more detailed analysis of the desorption measurements showed that the desorption rate decreased with time and that over the 4-day period during which the above values were determined, a fourfold reduction in rate occurred. A correlation between deposition velocity and desorption rate was noted.

Megaw and May (*83*) exposed circular disks with different surfaces in the Pluto experiments to a low concentration of molecular iodine (0.35 $\mu g/m^3$) for 1 hr. After exposure the disks were placed in sealed plastic bags, removed from the reactor shell, and counted as soon as possible. Three disks of each surface type were exposed, and desorption rates were determined in three atmospheric environments: clean air, air containing about 1 $\mu g$ of stable iodine per $m^3$, and air containing about 3 g of stable iodine per $m^3$. Results are shown graphically in Fig. 4.11. It is clear that little desorption of $^{132}I$ from copper or charcoal surfaces occurred either in the presence or in the absence of airborne $^{127}I$. Varying degrees of desorption from the other surfaces were observed, and exchange of surface-bound $^{132}I$ with airborne $^{127}I$ increased both the rate and the extent of desorption. These results confirm the desorption results obtained by liberating $^{127}I$ in the reactor shell, as shown in Figs. 4.1 and 4.2.

Parker and his co-workers (*46*) have found that part of the iodine deposited on the dry stainless steel surface of a simulated containment shell is rather readily removed by passing a stream of argon or air through the vessel. This is shown by data in Table 4.5, discussed in Sect. 4.2.2. In another experiment, molecular iodine (approximate initial concentration, 2 $mg/m^3$) was allowed to deposit for about 4 hr in the steel vessel filled with air at 15 psig (relative humidity, approximately 60%). Reduction of the air pressure at the end of this period from 15 to 2.5 psi removed about 2.8% of the iodine inventory, while sweeping with $2\frac{3}{4}$ tank volumes of argon removed 7.0% and $2\frac{1}{2}$ tank volumes of air removed an additional 2.9%. Corresponding figures obtained after aging the iodine aerosol for 18 hr in a steam-air mixture were 2.5, 2.4, and 1.7% respectively. Higher removal values were obtained with shorter exposures and where organic impurities were present in the vessel atmosphere. Fairman (*104*) reported losses of 32 to 39% of the $^{131}I$ during three weeks' exposure to air, from plates prepared from water solutions, using plates on which the iodine was deposited as AgI as a reference for 100% retention. It is obvious that care must be exercised in the handling of deposition coupons and other experimental devices having adsorbed iodine in order to avoid losses.

Collins *et al.* (*48*) reported that comparatively large (several percent of the iodine reaching the charcoal trap) amounts of penetrating iodine,

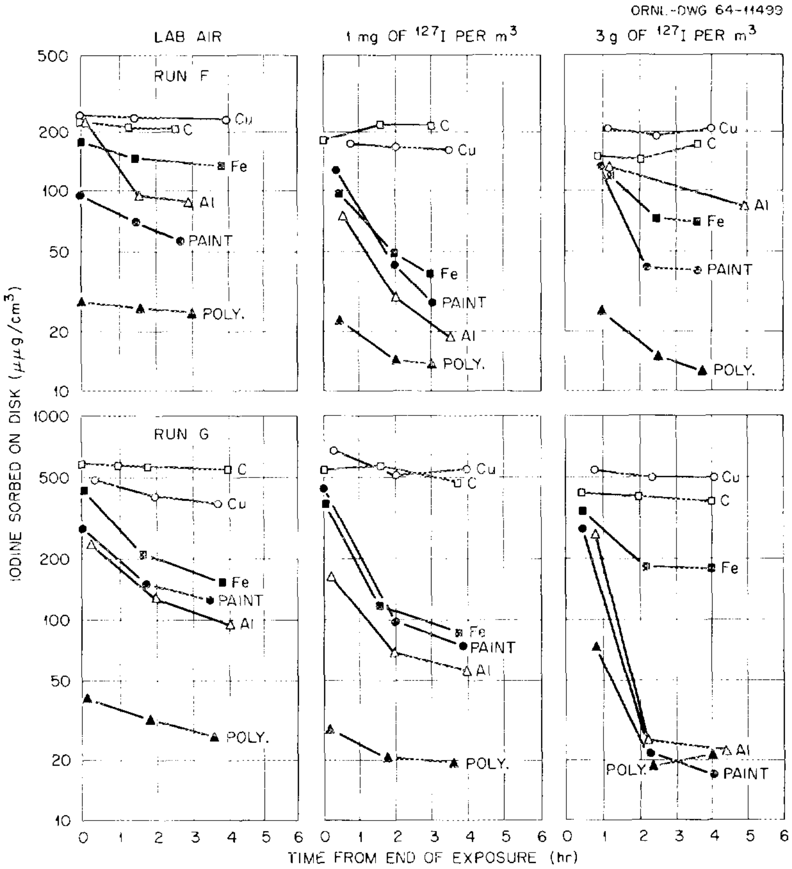


Fig. 4.11. Desorption of Iodine from Disks Exposed in Pluto (*81*).

identified as a mixture of alkyl iodides, were formed in an experiment involving release of iodine from irradiated $UO_2$ in $CO_2$ at temperatures in the range 1500 to 1700°C. A large fraction of the iodine released from the fuel deposited on vessel walls and other surfaces at temperatures up to 400°C. The authors state that it has been shown in other experiments (undocumented) that iodine desorbed from surfaces contains an enhanced proportion of alkyl iodides. This observation could have an important effect on the design of trapping systems for post-accident trapping of released iodine.

### 4.4.3 Desorption in Air at Elevated Temperatures

Two desorption experiments were carried out by Croft and co-workers (*91*). The first sets of deposition samples were heated in a small oven at temperatures of 50, 100, 150, and 200°C, for periods of 2, 4, and 24 hr, and the resultant desorption was measured after cooling. In the second experiment, some steel and concrete samples only were examined at temperatures approaching 300°C. Paint samples were not heated beyond 4 hr at 200°C, as under this condition the surfaces began to exhibit severe discoloration and flaking.

The results of the first desorption experiments are said (*91*) to exhibit certain main features, as follows:

1. The mean value $\phi$ decreased with the length of heating at a given temperature.

2. The mean value $\phi$ increased with temperature for a given length of heating;

$$\phi = \frac{1}{S_1} \frac{\Delta S}{\Delta t},$$

where

$\Delta S$ = loss in surface iodine concentration during $\Delta t$,

$S = S_1$ at $t = 0$.

3. Up to temperatures of ~150°C the relative desorption behavior of the specimens did not change significantly from that under ambient conditions, though the absolute rates increased.

4. PTFE shows the highest desorption factor for the three temperatures to which it was subjected. In contrast, the concrete samples exhibited the least facility for desorption during almost all the tests conducted. This was such that even for 24-hr periods at 200°C the resultant loss from the concrete was still only ~12%. For the painted samples a significant increase in desorption at temperatures approaching 200°C was associated with the observed surface deterioration.

Other steel and concrete samples were subjected to a different heat cycle, involving heating for ~20 min from ambient to a preset temperature, followed by natural cooling in air for ~10 min and subsequent recounting.

The main results of the second desorption experiment were as follows:

1. Up to a maximum cycle temperature approaching 300°C, the concrete samples nowhere showed a rapid increase in resultant desorption. For the metal specimens, however, there was a slow increase in percentage loss up to maximum cycle temperatures of ~225°C, by which time some 5% was lost per cycle, and between 225 and 250°C the onset of a rapid increase occurred, such that at ~270°C some 70% was lost per cycle.

2. Recycling of these metal specimens generally failed to reproduce this rapid increase, the rate remaining virtually independent of temperature after one cycle. One or two samples which had particularly high initial deposition levels showed a second increase at the same temperature, but of reduced proportions. It was also found that other mild steel samples, which had been heated for periods at lower temperatures, failed to exhibit the enhanced desorption at the higher temperature. The onset of the pronounced increase in desorption rate from the mild steel surfaces between 225 and 250°C probably results from a chemical process initiated at this temperature. It is known, for example, that the solid halides of iron sublime at relatively low temperatures, and some such reaction may be occurring here.

## 4.5 SUMMARY

A great deal of information on the transport behavior of iodine has been accumulated in recent years, but the data obtained serve better to indicate the need of further investigations than to supply definite answers to questions concerning the attenuation factor for iodine deposition in containment shells. One area that needs more attention is the role of submicron particles in iodine transport. It is clear that many factors affect iodine deposition behavior, but few, if any, of the parameters have been adequately evaluated under realistic accident conditions.

Large-scale tests in real or simulated containment systems have been made, in general, with unrealistic sources (molecular iodine liberated in the absence of vaporized fuel or cladding particles). Small-scale experiments with more realistic sources and with atmospheres currently of greatest interest in the U.S. nuclear safety program (steam and steam-air mixtures) have been made, to date, only with stainless steel simulated containment shells. Much useful data are being produced in experiments of this type, but more meaningful deposition rate data can be obtained by use of surfaces more nearly representative of the walls of real reactor shells, such as painted mild steel. Facilities that will permit parametric studies of deposition rates of iodine and other fission products on such surfaces are in operation at Hanford, and similar equipment is presently being designed at ORNL. Information coming from new and from existing installations should markedly reduce the areas of ignorance in iodine transport behavior in containment systems within the next two years. Data obtained under similar conditions with simulated reactor shells of increasing size will aid evaluation of the surface-to-volume ratio parameter and permit extrapolation of results to full-scale reactors. Results of full-scale reactor meltdown experiments in the LOFT facility will, eventually, lend more confidence to the results of the smaller-scale tests.

# 5. TRAPPING OF FISSION-PRODUCT IODINE

The consequences of a reactor accident can be significantly reduced by removal of released radioactive fission products from gases, before they escape to the environment. Accordingly, development and testing have been carried out on gas cleaning methods for this application, making use of particulate filters, adsorbers, scrubbers, foam encapsulation, steam condensation, scavenging, diffusion boards, and pressure suppression designs. This general subject has been reviewed in *Nuclear Safety* (*99*), and a series of conferences has been held on the subject of gas cleaning under sponsorship of the USAEC (*105–109*).

The design of a system for removing radioactive fission products from a gas depends upon the physical and chemical forms of the fission products and upon the nature of the gas. They may appear condensed as an aerosol or as gaseous impurities having a variety of chemical forms. Iodine is of special interest because of its importance as a potential contaminant in accidents and because of the previously discussed chemical variability (*25, 40, 67, 110–123*). To maintain complete control over fission products released in an accident, one must provide gas cleaning devices which are effective against each of these particulate and gaseous materials and must know the efficiencies of these devices.

The removal of radioiodine from gases has been the subject of a series of extensive reviews (*124–128*) which covered removal by a variety of materials. Materials used included activated charcoal, beds of silver-plated wire or heated silver nitrate, silver or copper mesh, caustic scrubbers, and dry soda lime. Forms of radioiodine studied included molecular iodine, iodine compounds, and iodine on particulates. Selection, design, testing, and efficiency of removal systems have also been reviewed.

Recent literature (*18, 81, 169*) contains information on the iodine trapping efficiency of these materials, comparing the effect of variations in iodine concentration, gas velocity, temperature, and the concentration of airborne impurities on the efficiencies for iodine removal from air, steam-air mixtures, helium, and carbon dioxide.

## 5.1 TRAPPING FISSION-PRODUCT IODINE

### 5.1.1 Particulate Filters

Elemental iodine is only partially removed by filters except when it is adsorbed on large particles. For example, a high-efficiency filter retained >99.99% of the cesium, tellurium, ruthenium, and uranium, as compared with 30% of the iodine released by melting stainless-steel-clad $UO_2$ in air (*25*). In an in-pile melting experiment in which a stainless-steel-clad $UO_2$ fuel element was melted in a helium atmosphere, a combination of roughing filter and HV-70 filter removed iodine with 92% efficiency (*40*). Particulate filter efficiencies in general ranged from 50 to 99% for iodine released from metallic uranium fuel elements heated to 1215°C in atmospheres of air, steam, or helium (*30*). In other tests where stainless-steel-clad dispersed-$UO_2$ fuel elements were melted in air, the released iodine was removed by a CWS-6 filter with 71% efficiency (*112*). In tests in which full-scale metallic fuel elements were melted in air and samples were collected at distances up to 5 miles downwind, iodine was the only fission product which penetrated absolute filters to an extent exceeding 5% (*129*). When aluminum-uranium alloy fuel elements were melted in dry air and in moist air atmospheres, the iodine was removed by the filter with 46% efficiency in dry air and 23% in moist air.

It is beneficial to have particulate filters in the systems for removal of the larger particles containing adsorbed iodine. The remaining iodine is then collected on solid adsorbents described in Sect. 5.1.3. An adaptation of charcoal paper filters has been used in sampling procedures (see Sect. 3.3.1). The activated charcoal filter material has been used primarily for collection of radioiodine near nuclear reactor installations. The paper is a combination of activated charcoal with a cellulose binder. Approximately 60% of the weight of the paper is composed of activated charcoal.

### 5.1.2 Liquid Scrubbers

J. M. Holmes (*128*) reviewed recent developments in iodine and ruthenium removal, the adsorption of nonradioactive contaminants from off-gas streams, and the design of several new cleanup systems. He emphasized that radioisotopes of iodine which contaminate the gaseous wastes from chemical processing plants are difficult to remove because of the various states in which they may appear in off-gas systems. For example, iodine can be present as iodine vapor; as a particulate solid, such as NaI; or as a solute dissolved in entrained liquid droplets or adsorbed on the surface of aerosol particles. A number of scrubber-type systems are reviewed as follows.

Taylor (*130*) studied the rates of iodine vapor absorption in water and in aqueous solutions of sodium hydroxide, sodium thiosulfate, sodium tetrathionate, and sodium sulfate. With a disk-type laboratory absorption column, the rates were found to be completely gas-phase controlled for the sodium hydroxide and sodium thiosulfate solutions; appreciable liquid-phase resistance to transfer was found with the water, sodium tetrathionate, and sodium sulfate absorbents. The gas-phase absorption rates were quite rapid, and a correlation with ammonia-water absorption data was obtained.

Several recent investigations emphasized the role of iodine adsorption on particles in reducing the efficiency of caustic scrubbers. May and Morris (*131*) conducted a series of iodine-absorption efficiency tests on the British BEPO reactor caustic scrubbers. The system comprised four 3-ft-diam towers, in parallel, containing 6-ft beds of 1-in. Raschig rings. A 5% sodium hydroxide solution was used to scrub 3000 cfm of air containing injected $^{131}I$. The predicted decontamination factor was 100 assuming perfect liquid distribution; this factor was 10 to 20 for the expected liquid distribution. The decontamination factors actually obtained for normal operation varied between 29 and 32. Substitution of sodium carbonate solution for the caustic did not change the efficiency. However, when particles were introduced into the inlet air stream, the decontamination factor dropped markedly. In one test, in which the air and iodine vapors were passed over a tungsten wire heated to 900 to 1000°C (a metal fume generator) before injection into the air feed, the decontamination factor dropped to 10.5. In another test the introduction of lead oxide particles reduced the decontamination factor to 1.4. The decrease in efficiency was attributed to the adsorption of iodine on particles that were not removed by the scrubber.

Chamberlain and Wiffen (*132*) demonstrated that iodine in air at a concentration of 10 $\mu g/m^3$ was rapidly and irreversibly adsorbed on lead fume but that the degree of adsorption on 0.1-$\mu$ condensation nuclei was small for the same iodine concentration. Experiments performed with the reactor containment shells and air cleaning systems of the British DIDO and PLUTO reactors (*133*) demonstrated the relative iodine removal efficiencies of a sodium carbonate scrubber, a copper knitmesh bed, and a charcoal bed in series with a Vokes filter. Only the activated charcoal bed–Vokes filter combination gave a reasonably high decontamination factor after the initial 1 $\mu g/m^3$ iodine concentration had dropped below 0.03 $\mu g/m^3$ as a result of adsorption on the shell and reactor walls. Since the Vokes filter (efficient down to 0.1-$\mu$ particles) did not improve the efficiency of the scrubber or copper bed, the adsorption of iodine on nuclei from the air was not considered to be the cause for their poor performance. The high efficiency of the charcoal bed led to the hypothesis that a gaseous compound of iodine was formed, perhaps from reaction of iodine with trace impurities in the air, and was strongly adsorbed on the charcoal but not removed by the scrubber or the copper bed.

### 5.1.3 Solid Sorbents

Radioiodine under most circumstances can be removed from gases most efficiently by activated charcoal adsorbers. The large surface area of activated charcoal has a rapid initial retention of elemental iodine. At room temperature the desorption is slight, possibly in the range of 0.001%. Adams and Browning (*114*) passed air through a 0.75-in.-deep bed of $^{131}I$-contaminated charcoal at a velocity of 75 fpm for 250 hr. The activity level was monitored periodically, and the decay rate was found to be identical to that of $^{131}I$ (8.05 days), indicating no loss of iodine at room temperature. At higher temperatures the iodine may become more mobile, particularly in the presence of oxygen, and there is a considerable difference in the performance of various types of charcoal. Chamberlain (*81*) quoted desorption data at 250°C in air ranging from 0.01% in 4 hr to 22.5% in 2 hr. The coconut-base charcoals have better retention but a lower ignition temperature. Adams and Browning (*134*) found an ignition temperature of 290°C for one type of activated charcoal in flowing oxygen. Chamberlain (*81*) reported ignition temperatures for various charcoals in flowing air of 300–500°C.

Charcoal-filter burning tests in air (*135*) indicate that a water spray activated by a temperature-sensitive device that has been considered for use in the trapping system of the HFIR at ORNL is effective in extinguishing charcoal fires. Carbon dioxide has also been used for this purpose in tests conducted by the Underwriters' Laboratories (*136*). When air flowed through a 20-in. by 20-in. by $\frac{1}{2}$-in.-thick bed at 200 lin ft/min, two 15-lb $CO_2$ extinguishers were required to cool and extinguish the fire. Approximately 25% of the charcoal was consumed. In general, filter systems are designed with enough natural cooling to prevent temperatures from reaching the combustion point of charcoal.

The presence of alkali metals in charcoal improves its retention, especially at high temperatures. The amount of iodine which a charcoal can retain at high temperature is limited by the number of metallic adsorption sites and also by the tendency of oxygen to replace iodine in metallic iodides at temperatures above 200°C. Possible substitutes for charcoal that would be less susceptible to oxidation effects are discussed in Sect. 5.1.5.

Nonelemental forms of iodine such as the organic iodides are adsorbed by charcoal, although with poor efficiency (see Sect. 5.1.4), but are not removed by silver, copper, or alkaline materials. This is believed to be one reason why copper, silver, and caustic scrubbers have shown poor performance in actual installations. The development of trapping methods for organic iodides is discussed in Sect. 5.1.4. Practical adsorbers have been designed having efficiencies of greater than 99.9% and 99.99% for molecular iodine (*111, 114, 137–141*). High efficiencies persist even after extended operation (*140–142*) and in the presence of steam (*142–144*). Radioiodine has been removed from helium at temperatures up to 700°C by activated charcoal loaded with metal salts (*145–146*). A combination particulate filter and charcoal adsorber collected 99.97% of the iodine released by melting aluminum-uranium alloy fuels in dry or moist air (see Sect. 5.3.4) and 98.6% of that released by melting stainless-steel-clad $UO_2$ fuel in air (*25*) (Sect. 5.3.5). One iodine vapor species was observed to penetrate activated charcoal under humid conditions in air, although it was effectively removed under dry conditions (*123*). Reduced efficiencies for adsorption of radioiodine by charcoal have been observed, especially at low concentrations, under special circumstances where the iodine had been exposed to impurities from chemical processing plants, metallurgical hot cells, or burning organic materials (*139, 141, 147–149*). Methods for testing iodine adsorbers in place after installation have been developed to ensure their continued effectiveness and are applied routinely to the adsorbers in the reactor compartment ventilation system of the NS "Savannah" (*144, 150–151*) (see Sects. 5.3.1 and 5.3.2). Similar tests were applied to the confinement system of the Oak Ridge Research Reactor (*152*).

Silver and silver compounds (e.g., $AgNO_3$ or AgI used as coatings) are very effective iodine removal agents. Removal efficiencies are as high as 99.9% in dynamic systems (gas flowing past materials such as Fiberfrax rings and metal screens coated with silver or a silver compound). Silver-coated materials have an optimum operating temperature of 190 to 220°C, but they work very well at 260°C. Water vapor has no effect on the removal efficiencies at 220°C for $AgNO_3$-coated materials, but the temperatures must be kept high enough to prevent condensation of water. Silver surfaces are reported (*114, 138–139, 153–154*) to be susceptible to interference by impurities or surface contaminations.

The chemistry of the reaction of iodine with silver compounds indicates that the equation

$$AgNO_3 + \tfrac{1}{2}I_2 \longrightarrow AgI + NO_2 + \tfrac{1}{2}O_2$$

is applicable, but at 190°C both silver iodide and iodate are formed. Silver iodide was stable in air at 350°C, but 78% decomposed at 550°C and 85% decomposed at 700°C in $\frac{1}{2}$ hr. Silver materials are potentially useful when placed ahead of a charcoal adsorber to reduce the generation of decay heat of radioiodine in the charcoal.

A clean reduced copper surface acts as a perfect "sink" for iodine at room temperature. However, it is difficult to maintain an unoxidized surface, and, for this reason, some investigators prefer to use silver-coated screens in May Packs rather than copper. Copper and copper alloys also effectively remove iodine from solution. Oxidized copper or copper oxide is relatively inefficient for iodine removal; efficiencies of 37.5, 4.3, and 0.5% were obtained at 125, 242, and 295°C respectively (*155*). Copper reacts with iodine below 242°C, but iodine is liberated from copper iodide above this temperature.

Morris *et al.* (*156*) made a thorough study of the adsorption and desorption of iodine in beds of copper; they also give some results of comparative tests with beds containing silver surfaces. Their data show that both materials are effective in removing molecular iodine from air streams.

Glass beads and sea sand are very inefficient for iodine removal. Lead will remove most of the iodine from a solution. Glass, stainless steel, and Lucite will remove a small amount from solution; aluminum appears to remove a slightly greater amount.

Adsorption of iodine on mixtures of silica gel and alumina at 106°C follows a BET theory (*157*) for adsorption on the dual surfaces, with alumina giving a type II and silica gel a type III adsorption. Molecular sieve material showed 99.9% or better (*170–171*) efficiency for iodine retention, with 55% retained in the first $\frac{3}{8}$ in. Iodine may be eluted from this material by a continued flow of air, unlike iodine adsorbed on charcoal.

### 5.1.4 Trapping Methyl Iodide

Collins and Eggleton (*158*) investigated the adsorption of labeled methyl iodide on coconut charcoal at different temperatures over a range of loadings. At 100°C quantities smaller than 100 μg were permanently adsorbed on a 25-g bed, but increasing penetration was observed as the loading was increased above this limit. At room temperature there was a longer delay before breakthrough occurred, due to enhanced physical adsorption, but the amount of methyl iodide permanently retained was considerably less.

The main duties of the trapping system of an advanced gas-cooled reactor are either to depressurize the pressure circuit after a single channel meltout or to provide continuous treatment of relatively clean coolant after a pressure circuit failure. The trapping system is not therefore required to operate for long periods while retaining a large amount of methyl iodide, and the critical loading at 100°C, which corresponds to 4 g of methyl iodide per ton of charcoal, would not be unduly restrictive.

Twenty or so common aqueous reagents, including strong oxidizing and reducing agents, soluble silver salts, and amines, were brought into contact with a gas mixture containing methyl iodide in a foam column for a period of 10 sec. None of them produced any worthwhile decontamination at room temperature except acid potassium bromate solution, which allowed a penetration of 1 part in 100.

Copper- and silver-impregnated charcoal showed no improvement over the standard coconut charcoal, but a coal-based charcoal (Sutcliffe Speakman and Co., Ltd., type 207B) was found to have a much higher loading limit. At 100°C a 25-g bed loaded with 300 μg of methyl iodide retained all but 2 parts in $10^5$ for 12 hr; loaded with 2 mg, the bed retained all but 1 part in $10^6$ for 1 hr.

The coal-based charcoal (type 207B) maintained its improved performance down to room temperature. Not only was a penetration as low as 1 part in $10^4$ observed in irradiated fuel experiments, but even when a 25-g bed was loaded with 13 mg of methyl iodide, less than 1 part in $10^4$ penetrated in 24 hr of elution. This loading, equivalent to 500 g/ton, may be taken as the permissible loading for reactor purposes.

The results obtained by Collins and Eggleton, discussed above, would seem to indicate that methyl iodide is adequately retained by coal-based charcoal either at 100°C or at room temperature. However, data obtained under different conditions by other investigators are less encouraging. May (results quoted by Chamberlain, ref. *81*) found more movement of iodine on a bed of 207B charcoal swept for 3 to 5 days with air, $CO_2$, and CO at 250°C or with $CO_2$ at 200°C than occurred in a bed of coconut-based charcoal under similar conditions at 200°C. McCormack (*159*) tested three (unspecified) grades of charcoal with 50- to 100-μg quantities of $^{131}$I-tagged methyl iodide, presumably at room temperature, and found that the iodine activity was rather quickly removed when air was passed through the beds. An elution half-time of 6 to 8 hr was found.

More significantly, Adams and Browning (unpublished data) have tested a sample of 207B charcoal under wet and dry conditions and have found a marked difference in retention efficiency.

The iodine source used in these tests was prepared by passing air over molecular iodine (made by the $PdI_2$ method) at $-78°C$ for 10 min with a flow rate of 30 $cm^3/min$. The iodine removed under these conditions was shown by diffusion tube measurements to be predominantly in the form of methyl iodide. This material was trapped in an approximately $1\frac{3}{4}$-in.-deep charcoal bed, and the bed was subsequently flushed with room temperature air at a velocity of 30–35 lin ft/min for 5 hr. When the air was dry, the 207B charcoal had an efficiency of 99.99%. However, when the humidity was raised to about 70%, the iodine retention efficiency under otherwise identical conditions dropped to 74%. A coconut-based charcoal tested with moist air under the same conditions (*160*) showed 97.3% efficiency. Collins and Eggleton's tests (*158*) were apparently made with a dry 95% $CO_2$–5% CO gas mixture. It appears, therefore, that 207B charcoal does not provide a solution to the methyl iodide under most probable accident conditions, where moisture will be present.

Collins *et al.* (*48*) reported that a bed of Hopcalite, a commercial preparation containing $MnO_2$ and CuO, retained about 90% of the methyl iodide presented to it, with a bed temperature of 100°C, and that most of the iodine penetrating the Hopcalite bed was molecular iodine which was efficiently trapped by a charcoal bed at room temperature. About 99.2% of the methyl iodide was retained by the two beds in series. Reduction of the efficiency of the Hopcalite resulted from the presence of the reducing gas, CO, which discouraged use of this material in $CO_2$-cooled reactor systems. Tests with this material for the retention of organic iodides in moist or steam-laden air are being planned at ORNL.

Parker *et al.* (*46*) mention use of a heated (620°C) bed of platinized alumina to oxidize organic iodides, backed up by a room-temperature charcoal bed to collect molecular iodine. The efficiency of this device and its optimum operating conditions remain to be determined.

### 5.1.5 High-Temperature Adsorbers

The efficiency of charcoal for trapping molecular iodine has been well demonstrated, as indicated by the discussion in Sect. 5.1.2 above, but concern has been expressed about the possibility of ignition of charcoal beds by exposure to high-temperature oxidizing gases during a reactor accident. It would be desirable, therefore, to have alternate extraction materials available that would be less susceptible to destruction under such conditions. Preliminary experiments were performed by Barton *et al.* (*161*) to test two different approaches to the problem: the use of activated alumina containing about 0.5% platinum (a hydroforming catalyst), and the use of iodide salts as a chemical exchange medium for gaseous radioiodine.

It was found that when a 3-in.-deep bed of granular KI was exposed to flowing helium carrying molecular iodine tagged with $^{131}I$ at 500°C, 53% of the trapped iodine remained on the bed after exposure to flowing helium at 500°C for 3 hr and at 600°C for 4 hr. The gas velocity was 10 lin ft/min throughout the experiment. These experiments demonstrated that exchange occurred between the gasborne $^{131}I$ and $^{127}I$ in the crystalline KI at 500°C and that potassium iodide beds are reasonably stable in flowing air at temperatures up to 600°C. However, this approach to the iodine trapping problem is not considered promising, because it would probably be necessary to maintain the bed at a temperature of the order of 250–300°C in order to guarantee efficient trapping of iodine by this material.

Tests with beds of platinized alumina showed relatively little movement of iodine in flowing air at 325 or 425°C, but some loss occurred in 1 hr at 525°C and almost 100% elution occurred during a 1-hr exposure at 625°C. In other experiments (*161*) the retention capacity of platinized alumina was compared with that of unplatinized alumina, using fission-product iodine produced by heating irradiated Al-U alloy specimens. It was found that the platinized material had a significantly higher retention capacity but that it lost about 10% of the trapped iodine during a 30-min exposure to air at 300°C flowing at a velocity of 8 lin ft/min. It appears, therefore, that this material would serve only to delay the passage of iodine at temperatures above 300°C and as a heat sink. Cheaper materials could serve effectively for the latter purpose.

Board and Davies (*162*) reported that activated alumina adsorbed 99.76% of the iodine carried by a stream of dry $CO_2$ with a bed temperature of 18°C and a flow rate of 40 ft/min. Under the same conditions, except for a bed temperature of 200°C, the efficiency was reported to be 99.56%. Data on the iodine absorption capacity of this material under wet conditions were not reported. It seems likely that saturation by water vapor would markedly reduce its ability to retain iodine efficiently.

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