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TECHNICAL PROGRESS REVIEW

NUCLEAR SAFETY

VOLUME 4, NO. 3

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A draft copy of Nuclear Safety, Vol. 4, No. 3, is being transmitted to each Technical Information Panel member, as well as to others concerned with the review for patent, classification, and administrative matters. If you have any comments or criticisms concerning the contents of this issue, please send copies to Mr. Thomas W. Hazard of DTI and Mr. H. F. Carroll of Declassification and to me. Inasmuch as the final draft of this issue is scheduled to go to press early in March, your comments will have to be received by February 15 in order to be considered for inclusion in the final publication. Mr. Hazard's and Mr. Carroll's addresses are as follows:

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Yours truly,

Wm. B. Cottrell by AWS

Wm. B. Cottrell
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TECHNICAL PROGRESS REVIEW

NUCLEAR SAFETY

Volume 4, No. 3

Prepared for U.S. Atomic Energy Commission
by Oak Ridge National Laboratory

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
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FOREWORD

This quarterly journal is one of a series of Technical Progress Reviews prepared by the Oak Ridge National Laboratory at the request of the Division of Information Services, U.S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. Nuclear Safety is not a comprehensive abstract of all literature published in this field during a given quarter, but rather a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered the province of health physics. Even with these exclusions, nuclear safety cuts across such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

Three distinctly different types of articles may be found in this issue of Nuclear Safety. In addition to the usual reviews of current

literature and special review articles on specific topics, this issue contains the second of a new series of feature articles. The feature article is a solicited monograph by a nationally recognized expert on particular topics of interest selected by the editors.

The special articles permit discussion of pertinent subjects that cannot be adequately considered by reference to only the current literature. The current review articles, however, constitute the major portion of this publication. All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. With the possible exception of the invited articles, interpretations in any article represent the opinions of the editors, who are employees of the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information.

It is recognized that the critical evaluation of subject areas leading to the determination of criteria cannot fail to stimulate contrary opinions. This is expected to be particularly true in the area of nuclear safety, since in many instances only preliminary information is available, the ramifications are many and varied, and opinion and judgment must be relied upon so heavily. While the editors do not propose that the pages of Nuclear Safety act as a clearing house for safety correspondence because of the above facts, we have had for some time a policy which would permit the publication of statements of position at variance with those expressed by the editors. Such statements will be published after

the editors have ascertained that a real difference exists and that the position is reasonable.

In addition to the invited contributors, many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

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FEATURE ARTICLES

THE RELIABILITY OF REACTOR CONTROL SYSTEMS

By E. Siddall*
Atomic Energy of Canada Limited
Chalk River, Ontario

(Editor's Note: The evolution of nuclear reactor technology, especially in the field of power reactors, places an ever-increasing demand on the performance of control systems in order that the plant design performance may be achieved. It is probably only natural that interest in control systems should increase simultaneously with the design of more advanced reactor concepts. In this article, Mr. Siddall describes the Canadian experience with control systems and outlines various means for improving the reliability of these systems. He has adeptly pointed up the importance of the organization of and communication between the groups engaged in the design, manufacture, installation, and operation of control systems. The phrase "control systems" as used herein is generic in the sense that safety and control functions are both included in the discussion.)

All large nuclear reactors make use of numbers of "control systems," which may be defined as assemblages of equipment intended for the adjustment of static and dynamic parameters in a larger process, and which usually cover startup, normal operation, and shutdown. They are, in general, concerned with the flow of information rather than the flow of energy, and they include instrumentation, monitors, alarms, and trip or scram systems. Devices used in typical control systems include transducers, indicators, recorders, amplifiers, relays, controllers, and a variety of output devices, such as reactivity control mechanisms and control valves.

Originally, process control was almost entirely a human function, but the growth of understanding of control has largely arisen from the

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study of automatic systems. At the present time, there is a universal trend for human control to be replaced by automation, particularly where continuous action or fast response are of value. The human being is left to contribute at a higher level in the control hierarchy and in a more deliberate manner.

Economics

Control systems are used to enforce operation of the process in the desired manner despite uncertainty, unsuitability, variability, and instability in the inherent characteristics of the process and its principal components. Their ever growing use and importance show that there is a net economic gain in their use. The factors involved in the net worth of a control system are listed in Table 1. The economic picture and the restraints that arise therefrom may clearly vary greatly between different control systems in one nuclear plant and between those in plants of different type and size.

Reliability

Control systems are not perfect in practice, any more than any of the other classes of equipment used in human industry. Failures of control systems tend, however, to have their main consequences in the main process, and usually a sharp distinction can be drawn between "safe failures," which tend merely to interrupt the profitable operation of the main process, and "unsafe failures," which tend to result in injury or damage. The propensities to be free from unsafe failures and safe failures are respectively defined as "safety" and "serviceability" for the

Table 1. THE TRUE ECONOMIC WORTH AND COST OF A CONTROL SYSTEM*

Debits	Credits
1. A fraction of the cost of maintaining the background of knowledge and a group of experienced people appropriate to the degree of technological advancement which the system represents; includes general research and development	1. That proportion of the revenue of the plant which can be credited to the control system because of its necessary contribution to the ordinary operation of the plant
2. The cost of designing and ordering the system; includes analysis, conceptual engineering, engineering in detail, and drafting, and also the administrative costs of obtaining and evaluating tenders	2. Any saving in capital and running costs in other parts of the plant resulting from the provision of the control system, including the value of any net saving in operating manpower
3. The cost of manufacture, including inspection and testing at all stages from raw material to the finished product	3. Any saving in the costs of repair, compensation, and lost revenue when the correct operation of the control system prevents accidents or reduces their consequences; includes the assessed humanitarian and political value of any reduction in real or fancied risk to human life
4. The cost of installation at the site, including supervision, special training of workmen, inspection and testing	4. Any salvage value of the system at the end of plant life; this is usually negligible but is included for completeness
5. The cost of commissioning the system so that it works as intended in relation to the whole plant	
6. The cost of evolutionary and remedial maintenance or replacement, including diagnosis of troubles and, in some cases, re-engineering of unsuccessful parts of the system	

Table 1. (continued)

Debits	Credits
7. The loss of revenue and cost of repairs to other equipment which may be attributed to malfunction of the control system	

*It is a definition of a successful control system that its net economic worth, when all factors are on a common basis, is positive. The credits and debits listed tend to occur at different times, and many can only be described a priori in terms of estimated probability. As in all economic studies, it is necessary to bring them to a common basis in time, for example, to an equivalent fixed charge at a specified date, or to equivalent annual charges over the life of the plant, before they can be added or compared.

purpose of this article, and the sum of the two is defined as "reliability." It may be seen from Table 1 that lack of reliability reduces the worth of a control system on several counts but that it is also not possible to call for an unlimited increase of knowledge and effort to make systems more reliable, since the cost of this must also be debited.

In nuclear plants, some of the control systems are of great importance economically and politically, since if their reliability is adequate, they can prevent a large fraction of possible nuclear accidents at the source, which is very much more satisfactory than merely palliating their consequences.

The salient features of the principal control systems in the Douglas Point plant, which is due to start operation in 1964 and which is probably typical of current designs in their broader aspects, are summarized in Table 2. Assessments of the relative importance of the systems to the over-all safety and serviceability of the plant are shown. It may be seen that measures to improve safety and serviceability may be quite selectively applied, and that is, in fact, what is being done.

Initial Methods of Increasing Reliability

It would be most misleading to assume that there is some "normal" level of reliability in control systems, since the degree of care and effort applied to their design, construction, use, and maintenance is obviously variable. For a given standard of engineering effort and with the use of the components typically available, however, the reliability of a system can be greatly affected by several specific measures taken in its design. Some of these measures that have received theoretical and practical attention in reactor control systems are the following:

Table 2. PRINCIPAL CONTROL SYSTEMS IN THE DOUGLAS POINT NUCLEAR POWER PLANT

System	Comparative Cost	Importance[a] in Respect to	
		Safety	Serviceability
Fuel channel inlet flow monitoring (306 channels)	$402,000	M	M
Fuel channel outlet temperature monitoring (306 channels)	202,000	M	M
Reactor flux-tilt control	23,000	M	M
Fuel failure monitoring	390,000	M	L
Moderator system controls	87,000	M	M
Primary coolant system controls	51,000	M	H
Primary coolant pressure control system	61,000	H	H
Shield-cooling controls	45,000	L	M
Liquid poison (shim) control system	34,000	L	L
Fueling-machine controls	260,000	M	L
Fuel-transfer system controls	46,000	L	L
Reactor regulating system[b]	182,000	H	H
Reactor protective (trip and set-back) system	59,000	H	H
Reactor building ventilation controls	35,000	L	M
Plant power controller	23,000	L	M
Turbine-generator controls	116,000	L	H
Turbine-generator automatic startup	23,000	L	L

[a]Takes into account the extent to which other systems cover the effects of failures; H = high, M = moderate, and L = low.

[b]The comparative cost of the regulating system excludes the cost of control absorbers and boosters, most of which are independent of control aspects.

1. The "fail-safe" principle. This is an attempt to improve safety by arranging for all the more common types of failure to cause the output of the system to revert to a known safe state, which, in the case of a reactor, usually consists of a shutdown. This measure increases safety at the expense of serviceability, with little or no effect on reliability.

2. Redundancy. This consists of the provision of more equipment than the minimum needed to carry out a function, usually in the form of parallel subpaths for the information flow. In general, if two paths are used, either safety or serviceability can be greatly improved. If three paths are provided, both safety and serviceability can be improved by taking the correct response to be that upon which at least two paths agree.

3. Periodic testing. In this approach, the correct performance of the system is established by periodic tests, which may or may not be more searching than is necessary. In a nonredundant system, it is only possible to do this by interrupting normal operation, and it is only possible to make the system fail safe if a fault is found. In redundant systems, however, such interruption is not necessary if only one subpath is tested at a time, and it is possible to continue operation via good paths while faulty ones are repaired. The periodic test is particularly valuable in systems having a binary output that normally stays in one state for long periods (for example, safe and unsafe), since a fault can readily mimic a normal condition and can persist for long periods if not revealed by testing.

Canadian Experience

Approximately nine reactor-years of experience involving some degree of application of the features listed above have now been accumulated in

Canada.[1] The NRU, NRX, and NPD reactors all have trip (scram) systems that are largely triplicated, with two out of three coincidence, and which can be tested with varying degrees of completeness during operation. They also have automatic power-regulating systems of considerable refinement, which cover the entire range of operation, except for occasional special conditions. These regulating systems are also largely triplicated. In NRX and NRU, agreeing channels are averaged to obtain the output. In NPD, broadly speaking, the channel giving an output signal intermediate in value between the other two (the "median signal") is acted upon. In all cases, if all three channels disagree to an appreciable extent, the reactor is shutdown; such a shutdown amounts to a fail-safe requirement. In NRX and NPD, the regulating system controls the level of the liquid (D_2O) moderator by means of valves in redundant arrangements. In NRU, a single system of movable neutron absorbers is used.

Many other systems of less importance are concerned in the operation of each reactor, and most of these are single path, with some use of the fail-safe principle and with periodic testing only as a maintenance procedure. The resulting mixture of systems, ranging from the very important ones that are fully triplicated, to minor ones of the simplest principle, has yielded valuable experience, the main points of which are discussed below:

1. Triplication improves reliability to a considerable degree and is now established for the more important functions; however, the improvement in serviceability is clearly much less than had been expected. This fact, as discussed below, has given rise to some new thinking. The actual improvement in safety is more difficult to assess, since truly

unsafe faults have always been rare in important systems both of earlier and later design. The most important unsafe condition in a regulating system occurred where the triplication was nullified by a design feature; this mistake is unlikely to be made in the future and must be charged to the learning process. In several other cases, single unsafe faults have been correctly absorbed in the triplication, and some were only detected by routine tests. There have also been incidents, however, where the redundancy was of little or no value, as explained below. On the whole, it seems to be accepted that a marked but not radical improvement in safety can be attributed to the proper use of redundancy and periodic testing. A very complete description and discussion of experience with the regulating systems of NRX and NRU has been issued.[2]

2. A minor feature of the way in which triplication has been applied in Canada, that is, the shutting down of the reactors on multiple disagreement of regulating channels, has degraded serviceability to a degree that is disproportionate to the small gain in safety.

3. The main reason for triplication being less effective than expected is that many of the troubles encountered did not meet the necessary condition of independence between different channels. Some important examples of trouble that arose during the commissioning and early operation of the NPD plant are listed below, with the basic problems underscored:

a. Lightning hit the power line and isolated the station. The voltage regulator was on manual control, so the loss of load caused a high voltage throughout the station. The control circuit fuses of two operating moderator-circulating pumps blew because

they were of too low a rating. The operator did not know how to start the third pump because switching was somewhat complex and that operator had missed the point during training. If the operator had been able to start the third pump, its control fuse would probably have blown also.

b. A considerable number of swaged-type tube fittings carrying heavy water at high pressure were found to be very badly made. They were all made by one contractor in racks assembled at his factory. The workmen making the joints had not been trained properly for the precision required. The trouble was not detected by the inspectors or by acceptance testing. Both organization and communication were at fault. The very few actual failures occurred at random times.

c. A considerable fraction of the differential pressure cells of one type was found to be subject to three different types of fault. One type was obvious and was therefore quickly remedied in the early stages of commissioning. The other two faults, which were design and manufacturing faults, occurred at random times after a period of operation.

It may be seen that redundancy had no value whatever in the first case.

4. The reason for the lack of independence between different channels is that many of the troubles encountered are of a different kind from those considered in most of the theoretical work on reliability. Instead of being in the nature of failures of components in systems that were in perfect condition immediately before the failure, troubles more usually consist of errors or weaknesses in concept, design, manufacture,

or installation that have previously escaped detection. Such defects can often be expected to affect all the equipment of one type, and the faults that are their symptoms may be precipitated by a common set of abnormal conditions.

5. The variety of troubles encountered is much greater than is usually considered. A large portion of faults defy any simple classification, and the whole story cannot be told in less than some hundreds of words. This greatly adds to the difficulty of publishing useful information on fault experience.

6. Many system failures cannot be corrected merely by the replacement of a component that has failed. Rather, the eradication of a more fundamental underlying design or construction error is necessary so that the chance of that type of failure occurring again is virtually eliminated. When knowledge of this type of fault is disseminated widely by formal and informal channels, remedial measures may be applied in other plants. This process accounts for the reported declining fault rate in the early life of equipment and appears to be of growing importance as better components and subsystems come to be used, for example, those used in NPD compared with the earlier reactors.

7. The different principles used, for example, mechanical, pneumatic, and electrical, do not result in noticeable differences in reliability of over-all systems when all the other variables are taken into account. There is more variation between the best and the worst examples of one principle than there is between principles. The replacement of aging thermionic tubes, usually before an actual system failure occurs, remains, however, a large maintenance task.

8. There is little evidence to show whether automatic operation is inherently more or less reliable than manual operation, mainly because so many other variables enter the comparison. There has, however, been an obvious growth of confidence in automation to the point where, for example, direct manual operation of reactors is now viewed as abnormal.

Potential Methods of Improving Reliability

In the light of this experience, it is expected that marked improvements will be made in the use of the previously listed features in the Canadian program. The basic control techniques and the equipment used should also improve considerably, if only because of the elimination of methods and devices found to be unsatisfactory.

On the philosophical side, however, three additional measures, not necessarily novel, appear to deserve recognition and attention:

1. Engineering. The achievement of reliability has resulted from improvements in the whole engineering process. Theoretical and practical treatments of system reliability usually include block diagrams of the system. A block diagram such as that shown in Fig. 1 appears to be most meaningful, however, in discussing troubles that have actually been experienced. All the troubles listed above, for instance, involve failures within particular blocks or in the communication channels between the blocks shown in Fig. 1. Effort will of course be required if the system shown in Fig. 1 is to be improved with respect to preventing failures within it from extending to the point of the operation of a reactor with latent defects in control systems. It seems likely, however, that this would be one of the most rewarding directions for effort.

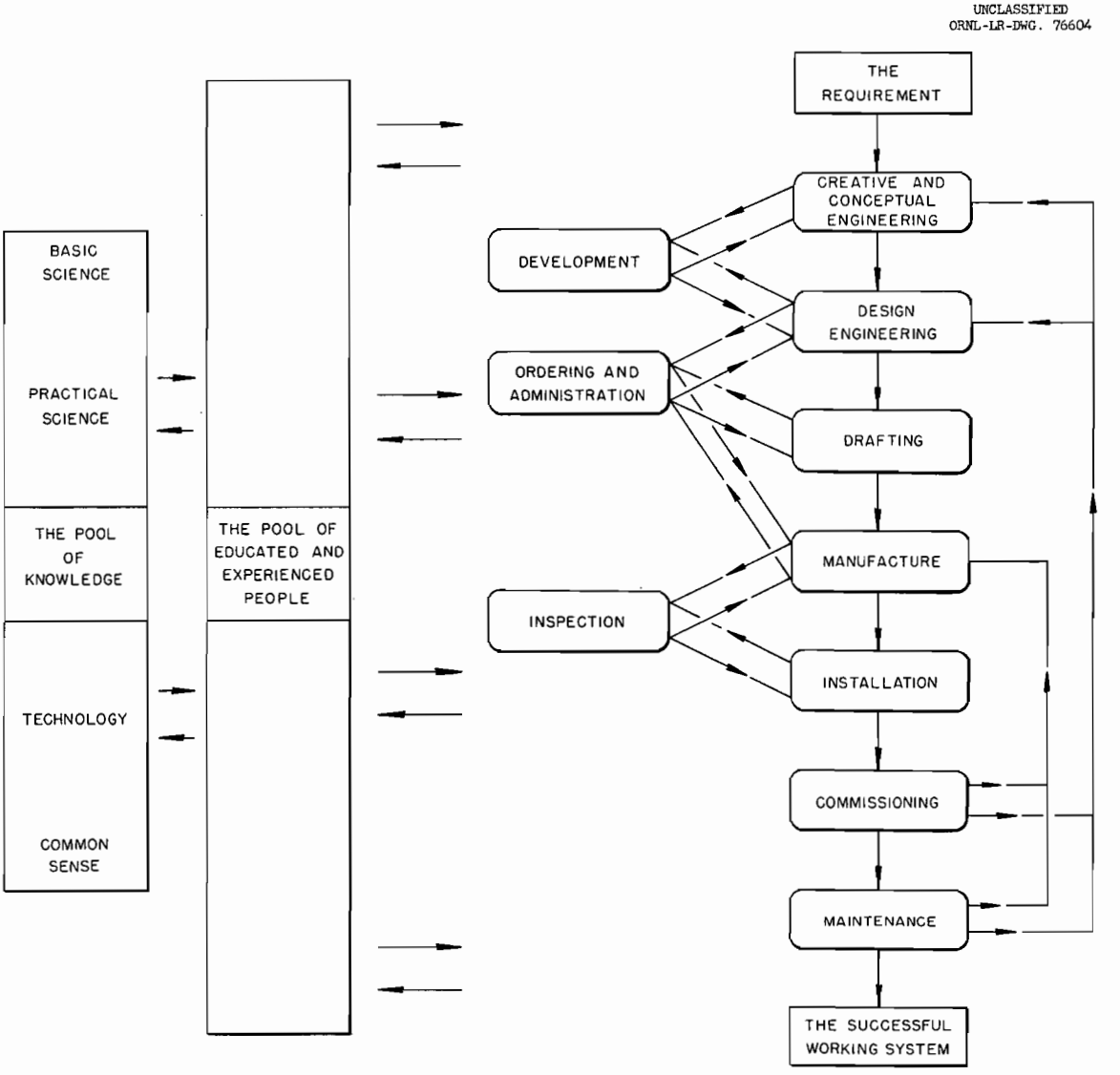

Fig. 1. The Engineering Process.

2. Flexibility. It will be noted that feedback paths are indicated in Fig. 1 that are based on development and field experience. These frequently imply changes in equipment after its original design and after it has actually been in service. Such evolutionary changes are extremely important to reliability, as explained above. This evolutionary process can be facilitated if equipment that can be conveniently changed is used. Reliability is also facilitated if standard techniques can be used as widely as possible so that the special problems of each application are not compounded unnecessarily by the use of different types of equipment for the same basic tasks. Such standard techniques must necessarily be very flexible if they are to cover a large fraction of the applications.

3. Underrating. For every device, limits must be placed on such parameters as the temperature, pressure, voltage, and current to which it is exposed or which it may be required to carry. In general, the higher such limits, the more the value and merit of the device. There is therefore a tendency for designers and manufacturers to set values for such limits that are as high as possible. There is, however, much evidence that such limits tend to be only statistical in nature and that, in fact, there is a continuous relationship between the value of each stressing parameter and the probability of failure in unit time. In particular, high temperature seems to be a universal cause of unreliability. It may well be that the inverse proportionality between the logarithm of the mean time to failure and the absolute temperature, which has been found to apply to good-quality solid-state devices, at least in some cases,[3] will be found to have general application. Underrating is now commonplace in electronics, but its importance does not seem to have

been recognized in other branches of control technology. Proper attention to the relationship between every working parameter and its supposed permissible limit, and to the basis of that limit, is essential for obtaining reliability in components.

Conclusion

The actual safety and serviceability, and thus reliability, that will be achieved in control systems in nuclear plants will depend broadly upon the relationship between the capability of the technology and the organization in which it resides, on the one hand, and, on the other hand, upon the demands which are made on the technology and organization in the form of complexity and number of systems which must be produced and used. If the proper philosophical and practical effort continues to be applied, there is every reason to expect that control systems will improve their dominant place in the safe and profitable operation of reactors.

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FAILURES IN HIGH-TEMPERATURE HIGH-PRESSURE PIPING

Helmut Thielsch*

Pipe is pipe! Its application in a commercial steam power plant, nuclear power plant, refinery, or chemical plant is usually based on identical or very similar design considerations. Materials of the same mechanical and physical properties, chemical compositions, and metallurgical structures are used. Fabrication procedures, such as bending, shaping, welding, and heat treatment, involve identical techniques, which depend upon the final quality desired and not on the type of application. Finally, inspection by radiographic, ultrasonic, or other techniques does not differentiate, for example, between nuclear and commercial power plant piping** applications.

Unlike steam power plants, which have been in operation for over half a centruy, nuclear power plants, in general, have been in service for only a few years. Relatively few piping failures have occurred to

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**The designation "piping" includes pipe, tubing, fittings, (elbows, tees, flanges, reducers, etc.), valves, headers, expansion joints, flow nozzles, and other components in the piping system.

date in nuclear power plants. Since piping systems in commercial steam power plants have usually been fabricated and welded to the most conservative specification requirements and have employed materials produced by the best steel-making practices, the experience with failures in commercial steam power plants may foretell some problems to be expected in the future in nuclear power plants.

As increasingly severe service requirements are being imposed on the materials used, service failures are becoming more frequent. A substantial number of steam power plants now have as much as 10 to 20% of their producing capacity idle because of service failures. These include not only piping failures, but also failures in boilers (boiler tubing), turbines, and other equipment essential to the safe operation of the power-producing unit.

The high-temperature high-pressure piping system of the type used in steam power plants are usually designed, fabricated, and welded to the requirements of the ASME Boiler and Pressure Vessel Code and the ASA Code for Pressure Piping. The materials specified have to meet, in general, the requirements of the specifications of the American Society for Testing Materials and, usually, the various supplementary test requirements listed. The piping components and systems are normally subject to various inspection procedures and to careful supervision during handling, processing, fabrication, and inspection.

To date the service requirements in nuclear power plants have involved lower temperatures than those of modern steam power plants. Nevertheless, service failures in nuclear plants are viewed with extreme concern, particularly where the leakage of radioactive materials might lead

to contamination and thus hazardous conditions for the replacement or the repair of the pipe section containing the failure. In extreme instances, a whole section, building, or even area might become inaccessible because of leakage resulting in radiation levels in excess of safe limits.

Classes of Service Failures

Service failures in piping systems are usually associated with at least one of the following five factors:

1. Design.
2. Material selection and handling.
3. Manufacturing and shaping of materials by the mill.
4. Final fabrication and welding in the pipe fabrication plant or during erection.
5. Excessively severe service conditions.

In some instances, a cause of failure can be related to a combination of several of these factors. For example, a pipe weld containing root defects, such as lack of penetration, may not fail during service until thermal or mechanical fatigue of unexpected severity causes propagation of the existing notch across the weld thickness.

Design Faults

Failures associated primarily with design may be the result of improper structural design, often involving insufficient flexibility. They may start at design notches, such as sharp corners, reinforcements, or attachments to the piping, which cause severe localized restraint. Failures also have started in welds where the weld-joint design made deposition of a sound weld difficult.

Thermal fatigue or shock or mechanical fatigue or shock have initiated and contributed to some of these types of failures. They often have propagated cracks across the pipe wall.

Piping System Design. A failure in a steam line, associated with insufficient flexibility, is shown in Fig. 1. During periodic shutdowns of the boiler, which caused cooling of the piping systems, thermal contraction occurred in the piping. Startup of the boiler, of course, caused thermal expansion again. Insufficient flexibility in the system to absorb the stresses resulted in the initiation of cracks that gradually propagated across the pipe wall. Such cracks occur most frequently near anchor points or major branch connections that act as restraints to the free movement of piping system components.

Design Notches. Where the design includes attachments for pipe hangers or supports, heavy welds along the ends may act as sufficiently severe restraint to initiate cracks, again as result of thermal or mechanical fatigue, as in Fig. 2. Failures have occurred where design conditions involved dissimilar wall thicknesses in castings, forgings, or welds joining heavy-walled valve bodies and pipe of lighter wall thickness. Socket welds and headers with reinforcing saddles also have failed in high-temperature high-pressure piping systems. Design engineers often do not realize the importance of gradual changes in section thickness.

Weld-Joint Design. Unfavorable weld-joint designs have resulted in a number of service failures. It is often not recognized that the primary concern in weld-joint design should be accessibility for the welder whose task it is to produce sound welds. Some weld joints are designed so that they are either difficult to produce under shop or field-erection environments or are susceptible to defect conditions.

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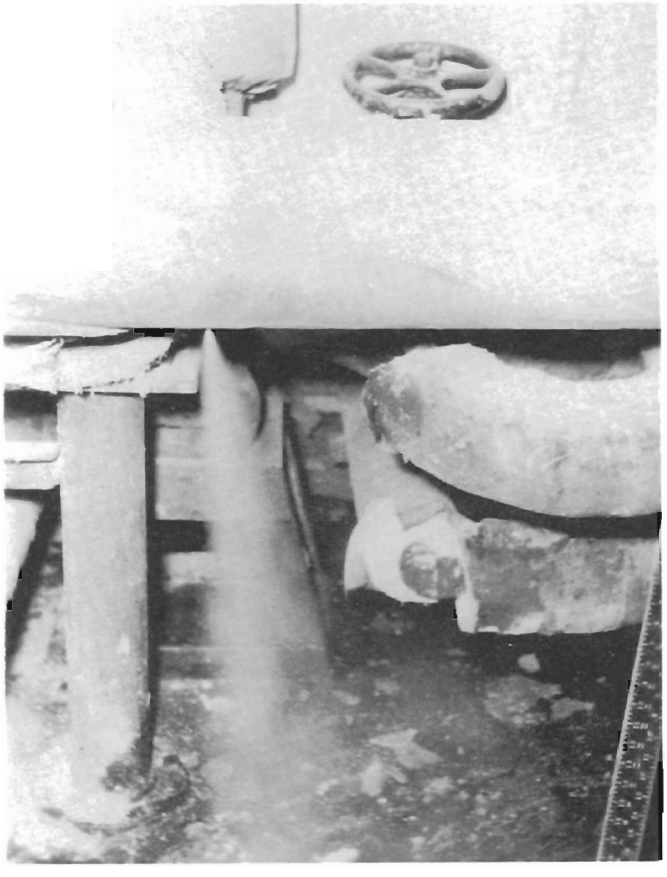

Fig. 1. Failure in Steam Piping Associated with Insufficient Flexibility in the Piping System.

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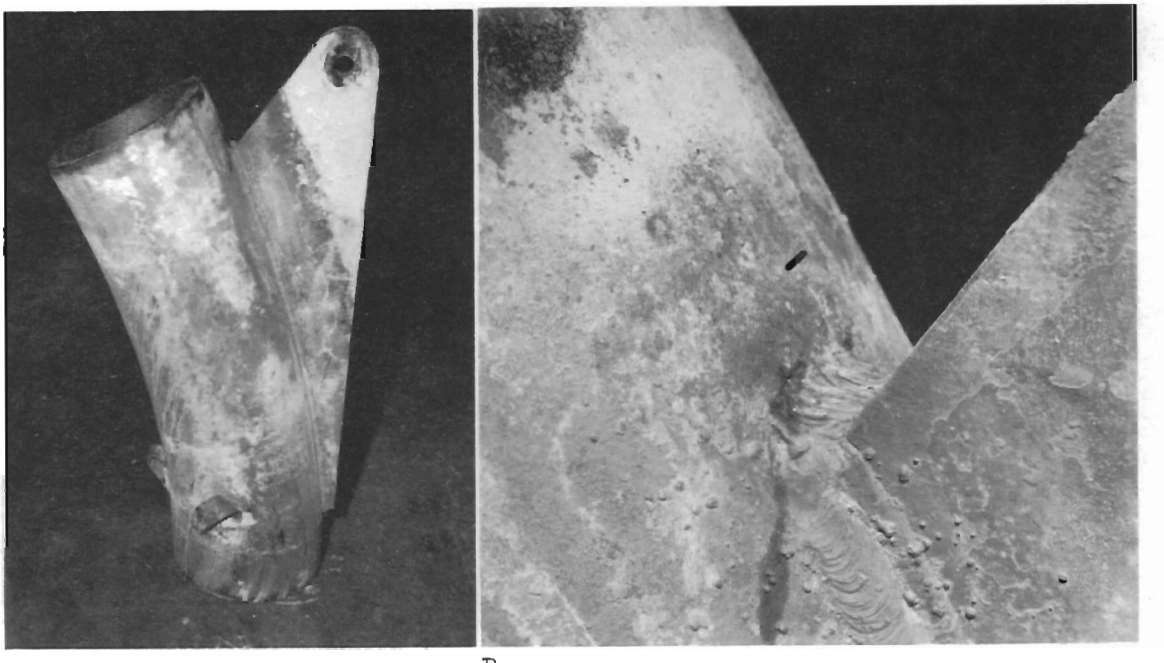

A B

Fig. 2. Failure by Thermal Fatigue at Welded Hanger Attachment Because of Sharp Change in Weldment Design. (A) Reduced to 1/20. (B) Reduced to 1/3.

Cost is also a factor. The 37.5-deg V-bevel, which is the cheapest to prepare and with which standard fittings are usually furnished, is not, in general, the best for critical piping systems. This is particularly true where the first weld pass is made by the inert-gas tungsten-arc welding process normally used in the fabrication of the piping systems of nuclear power plants. Since cracking in the first (root) weld pass is not always detected by radiographic examination (Fig. 3), it is very important that weld-joint designs be specified that provide the greatest assurance of weld soundness. In certain critical-use areas, an unproven joint design should be evaluated by tests that simulate, to a major degree, the service conditions. Low- and high-cycle fatigue and thermal-shock tests are desirable.[1] These tests may be helpful, also, in the evaluation of radiographic standards and permissible levels of porosity, tungsten inclusions, and other film indications of nonhomogeneities.

Material Selection

Materials for piping systems should be specified by selecting the most economical material that meets the requirements (1) of the service conditions and (2) of the applicable codes and specifications. Engineers who specify materials for high-temperature high-pressure piping systems, however, often do not give sufficient recognition to the effects of temperature, pressure, stress levels, restraints, corrosion, oxidation, section size and shape, thermal fatigue or shock, or mechanical fatigue. To play it safe, materials are sometimes selected that are far more costly than necessary. Failures resulting from improper welding or other defects may occur just as readily in the incorrectly used, more costly material as in the "proper" material.

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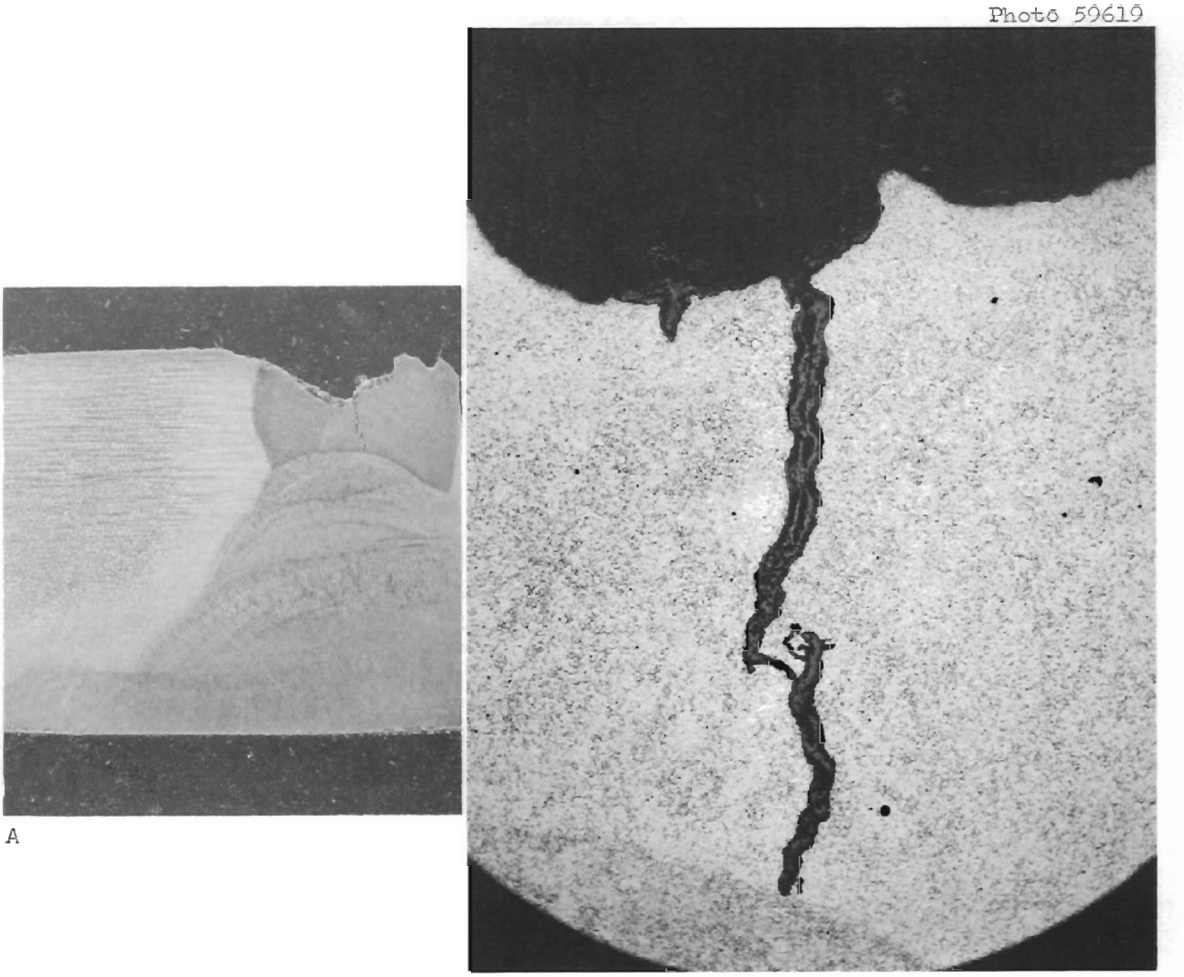

A

B

Fig. 3. Crack in Root Pass of Pipe Butt Weld with 37.5-deg V-Bevel Preparation. Crack was not detected by radiographic examination. (Magnifications, A = 8 diameters, B = 50 diameters.)

Pipe Base Metal. Selecting materials on the basis of laboratory creep, stress-rupture, fatigue, or corrosion tests has resulted in a number of failures. Piping with thick walls may be susceptible to failure in service environments, even though in accelerated high-temperature tests of small-scale laboratory specimens it exhibited good properties. Examples of such failures have been the well-publicized failures of carbon-molybdenum steel piping at temperatures over 800°F because of graphitization adjacent to weld deposits in the heat-affected zone (Fig. 4). Similarly, some type 347 stainless steels have failed at temperatures over 1050°F in the pipe base metal adjacent to weld deposits (Fig. 5).

Welding Filler Metals. Welding filler-metal selection and handling is equally important. Figures 6 and 7 illustrate cracks developing in a weld joint where a 1% chromium–0.5% molybdenum steel pipe has been welded with carbon–0.5% molybdenum steel electrodes. The weld joint had been in service for approximately two years at 1000°F. Carbon migrated from the weld deposit into the base metal and left a decarburized zone in the weld deposit adjacent to the base metal. The resulting differences in properties were of sufficient magnitude to cause the initiation and propagation of cracks during the high-temperature service involved. Failures in high-temperature service have occurred even in carbon-steel weld deposits adjacent to carbon-molybdenum steel piping where the difference in composition is only 0.5% molybdenum.

Manufacture and Shaping of Materials

Perfect piping materials are not made commercially. Defects may range from atom-size dislocations in the metal, which cannot even be

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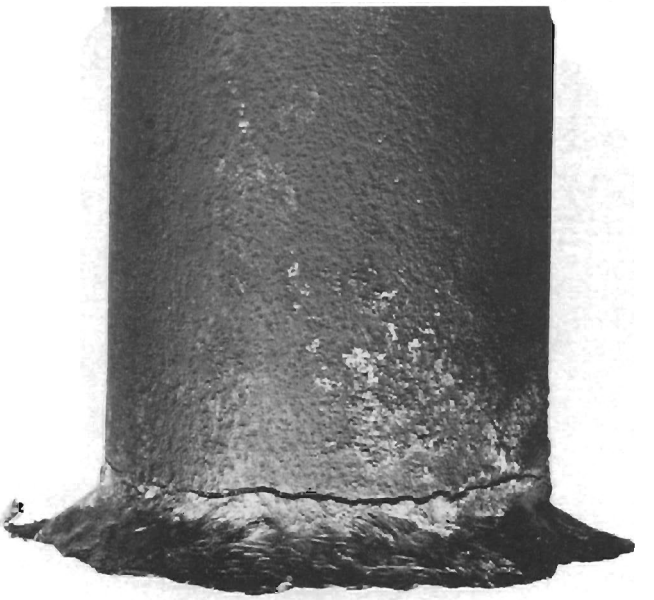

Fig. 4. Crack in Heat-Affected Zone in Carbon—0.5% Molybdenum Steel Piping Caused by Graphitization After Service at Temperatures Above 800°F.

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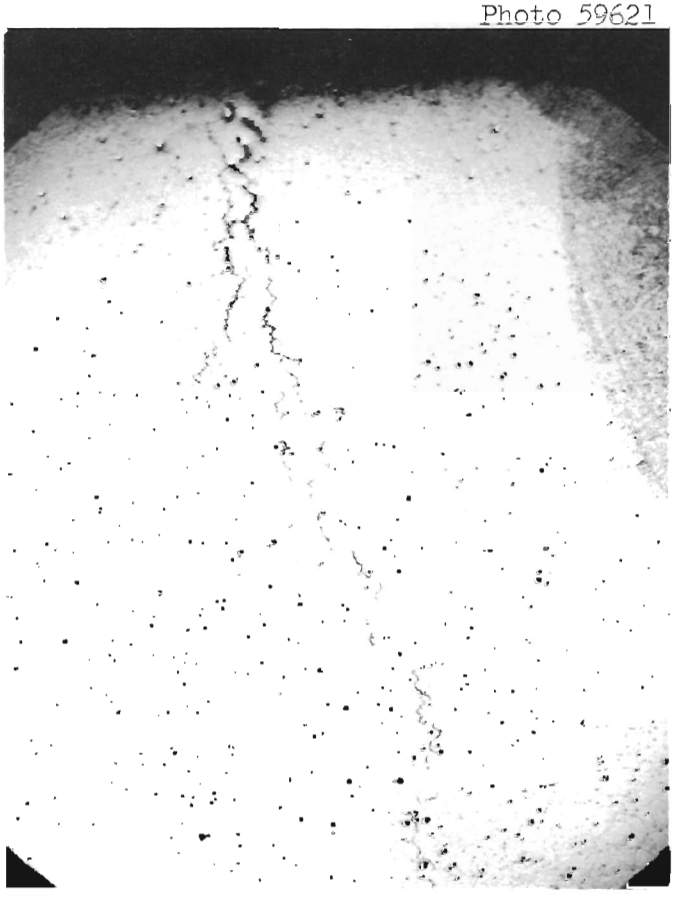

Fig. 5. Cracks in Heat-Affected Zone of Type 347 Stainless Steel Welded Joint that Occurred at Steam Temperatures over 1050°F. (Magnification, 35X)

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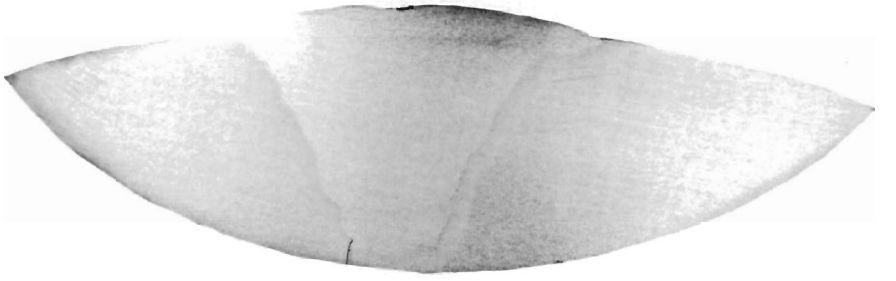
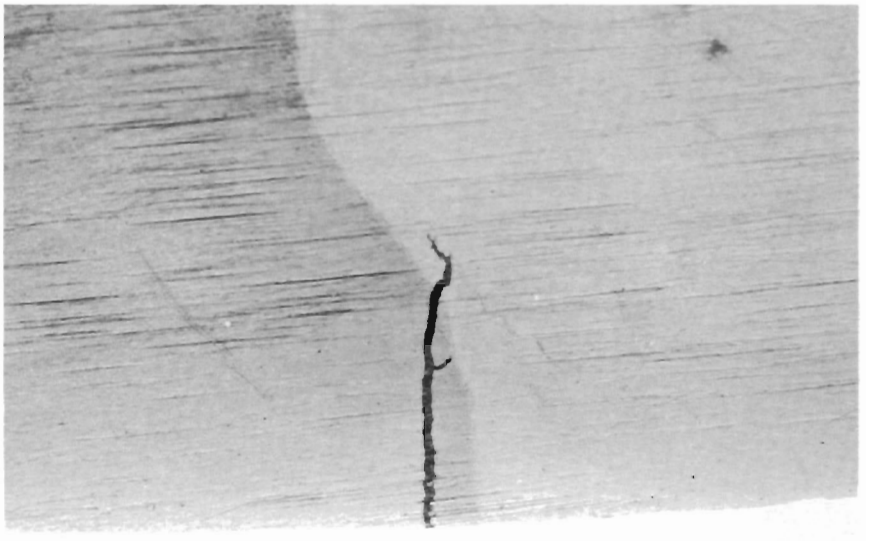


A



B

Fig. 6. Crack in a Welded Joint in a 1% Chromium–0.5% Molybdenum Pipe Welded with a Carbon–0.5% Molybdenum Steel Electrode. (Magnifications, A = 1X, B = 10X)

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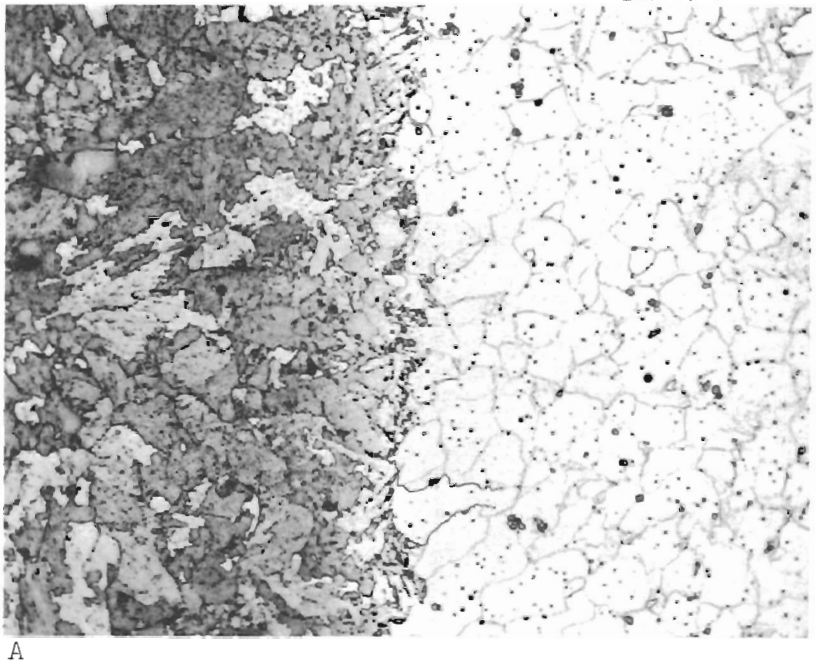

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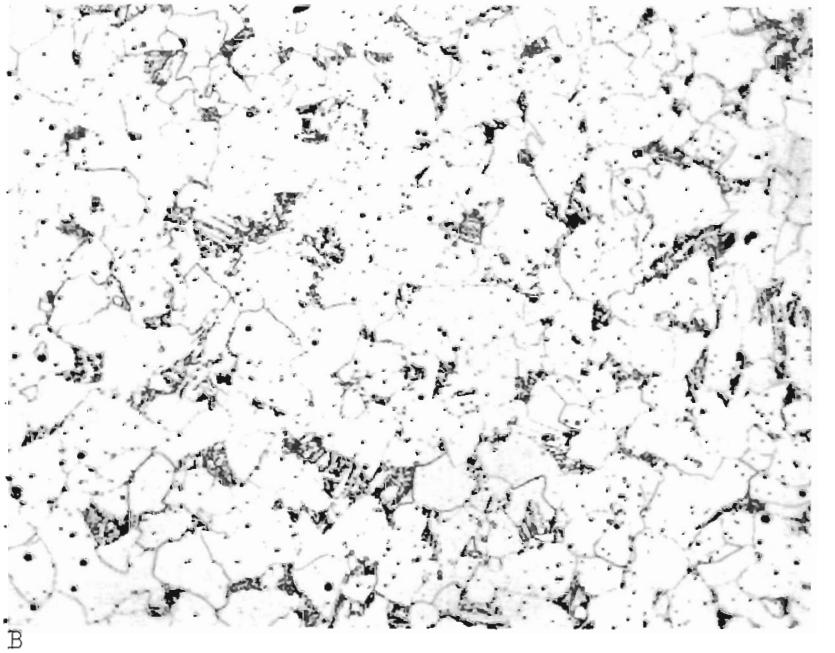

B

Fig. 7. Microstructure (A) of Bond Between Decarburized Carbon–0.5% Molybdenum Weld Metal and 1% Chromium–0.5% Molybdenum Base Metal and (B) of Weld Metal Away from Decarburized Zone. (A) Effects of carbon migration from weld deposits to base metal may be seen. (Magnification, 500X)

observed under the highest power microscopes, to major metal discontinuities visible to the eye. Some visible defects may not reduce the service life of the piping component; on the other hand, many visible surface defects and invisible subsurface defects can result in service failures.

The defects caused by the manufacture and shaping of materials for high-temperature applications may have been introduced into the casting at the ingot stage. Defects may have also been produced during the subsequent rolling, forming, reduction, or extrusion of the ingot or billet into the final pipe product or into elbows, tees, flanges, or other shapes. Failures for which the manufacturer of the high-temperature piping component is responsible usually involve (1) notch-type defects or (2) metallurgical notches.

Notch-Type Defects. Notch-type defects involve actual discontinuities or separations in the metal either at the surface or on the inside. Laminations, scabs, slivers, etc., represent very common types of metal separations in seam-welded or extruded steel piping. Laminations perpendicular or diagonal to the pipe surface should be considered as more critical notches than those parallel to the pipe surface. Shallow surface slivers and laminations parallel to the surface rarely have resulted in failures unless they were very heavy or reduced the wall thickness of the pipe substantially. A critical lamination on the inside of a stainless steel pipe for high-temperature service is shown in Fig. 8.

Mechanical notches caused by pipe or tube drawing, extruding, or machining have resulted in a number of failures of the type illustrated in Fig. 9, where cracks started from a draw-die groove on the inside of a tube. Mechanical fatigue initiated and propagated cracks at the notch

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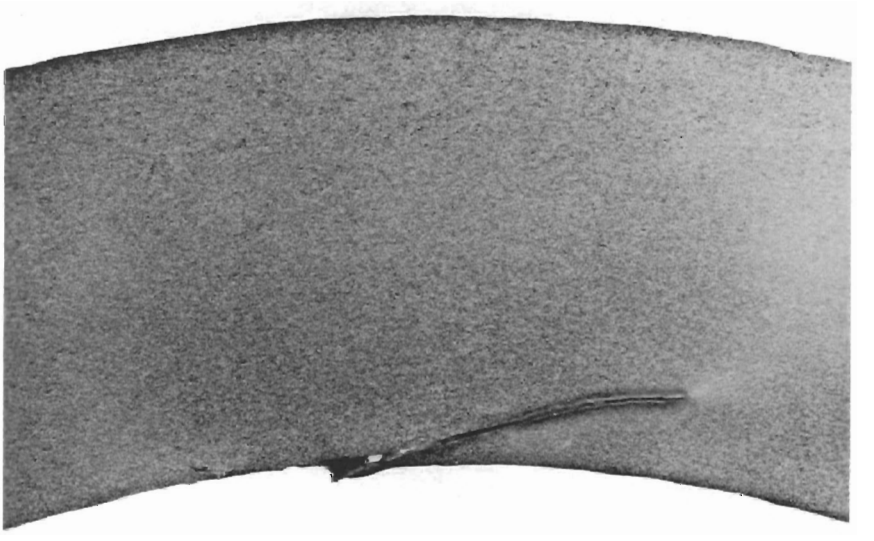

Fig. 8. A Critical Lamination on the Inside of a Stainless Steel Pipe Section for High-Temperature Service. (Magnification, 5X)

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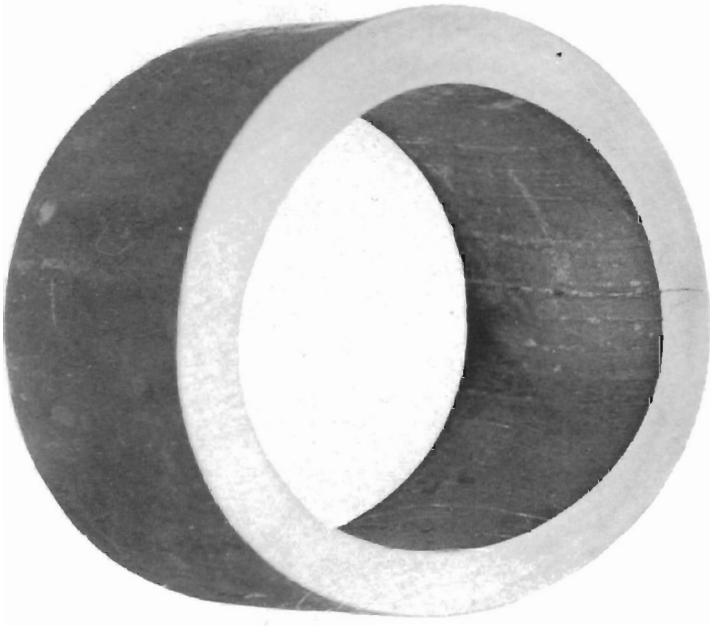

Fig. 9. Crack Starting at Notch Formed by Drawing-Die Groove.

formed by the die groove. Service environments involving stresses and corrosive conditions are even more likely to initiate cracks in such grooves (Fig. 10).

Internal defects are quite common in castings used in high-temperature piping systems as fittings, valve bodies, etc. Only in rare instances have internal gas pockets or shrinkage defects caused actual failures, and then they occurred close to the inner or outer surface, where propagation of the shrinkage into cracking was caused by mechanical or thermal fatigue.

Cracks in a carbon—0.5% molybdenum steel valve body are shown in Fig. 11. Shrinkage cavities present in the same valve body near the area where the cracks occurred did not contribute to the failure. The shrinkage cavities are illustrated in Fig. 12.

Metallurgical Notches. Metallurgical notches represent differences in properties of sufficient magnitude to become a potential cause of failure. Such conditions can occur within the same metal, where localized heat-treatment produces an area of high hardness adjacent to an area of lower hardness. Metallurgical notches also can represent a hardened heat-affected zone adjacent to a weld deposit, or the differences in properties between a wrought base metal and a weld deposit, which is essentially a casting. More apparent metallurgical notches occur between dissimilar metals.*

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*In welds between dissimilar base metals, the severity of the metallurgical notch may vary with the welding filler metal selected. Selection of the welding filler metal should include consideration of the coefficients of thermal expansion in the respective materials involved. Moreover, welding procedures and heat treatments must be carefully specified.

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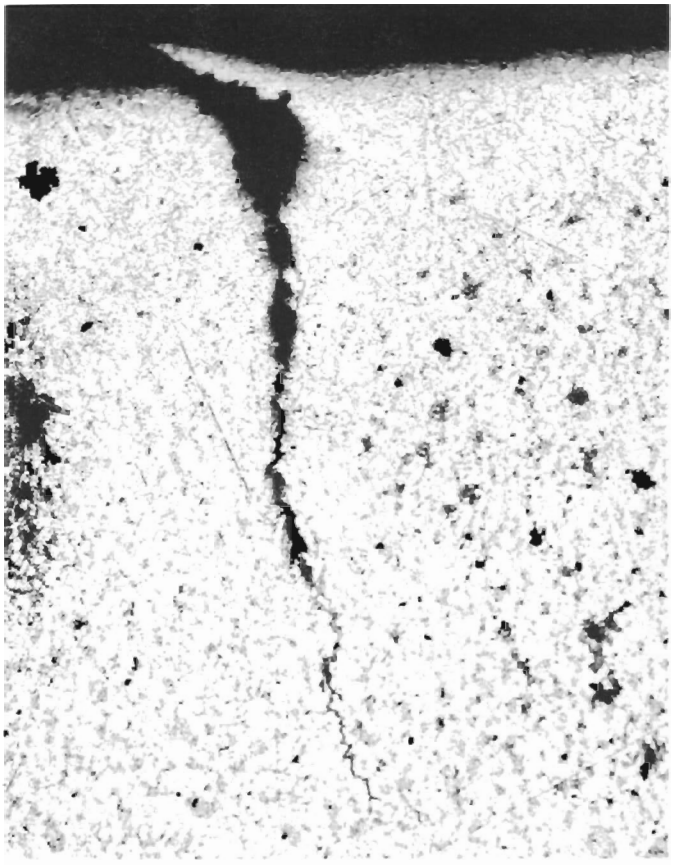

Fig. 10. Stress-Corrosion Cracking Initiated in Groove on Inner Surface of Steel Tube. (Magnification, 50X)

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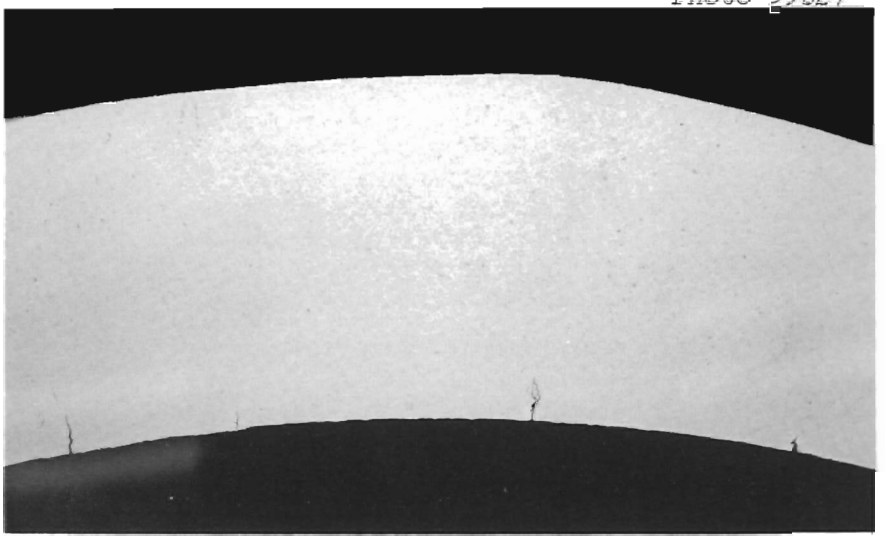

Fig. 11. Cracking in Carbon-0.5% Molybdenum Steel Valve Body Starting from Inner Surface. (Reduced to 9/10)

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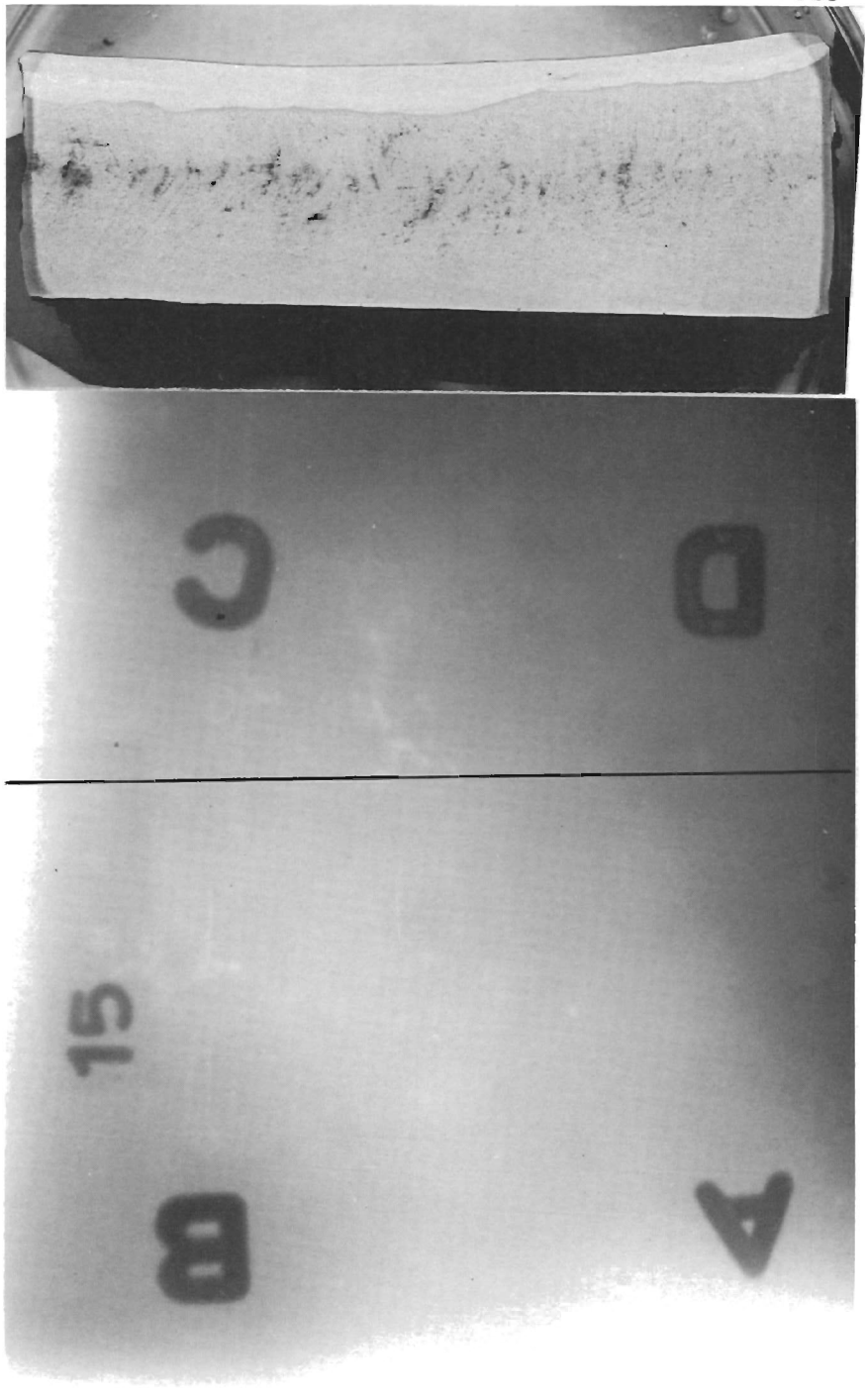

Fig. 12. Cross Section and Print of Radiograph Illustrating Shrinkage in Cast Carbon—0.5% Molybdenum Steel Valve Body with Weld Repair Overlay on Outer Surface. (Reduced to 9/10)

Failures have occurred where the differences in hardness exceeded approximately 50 to 100 points Brinell, and particularly where the piping system has experienced thermal or mechanical fatigue. A failure along the weld-to-base metal interface in a carbon–0.5% molybdenum main steam piping system is illustrated in Fig. 13. The base metal was a valve body of somewhat higher than normal carbon content (0.26–0.31%). As often occurs in castings, variations in the chemical composition may further intensify the susceptibility toward localized hardening. As was normal at the time of welding ten years ago, the weld used was stress relieved at approximately 1150°F. After over ten years of service at 900°F, the Brinell hardness values* were the following:

Valve base metal	165–189
Valve heat-affected zone	231–235
Valve heat-affected zone adjacent to weld	268–290
Weld deposit	177–181

A lower carbon content in the valve base metal or a higher heat treating temperature, which might now be recommended, would have resulted in a reduced hardness differential along the weld metal-to-valve base metal interface. This should have avoided this type of failure.

Carburization or decarburization of the steel along the inner or outer surfaces can also introduce metallurgical notches. Surface carburization in a seamless elbow of 1.25% chromium–0.5% molybdenum alloy steel, which is extensively used in high-temperature high-pressure piping, is shown in Fig. 14. Butt welds subsequently made on the elbow, which

*Hardness determinations were made with a microhardness tester providing Knoop hardness values. These were converted to the Brinell values more generally applied to steel materials.

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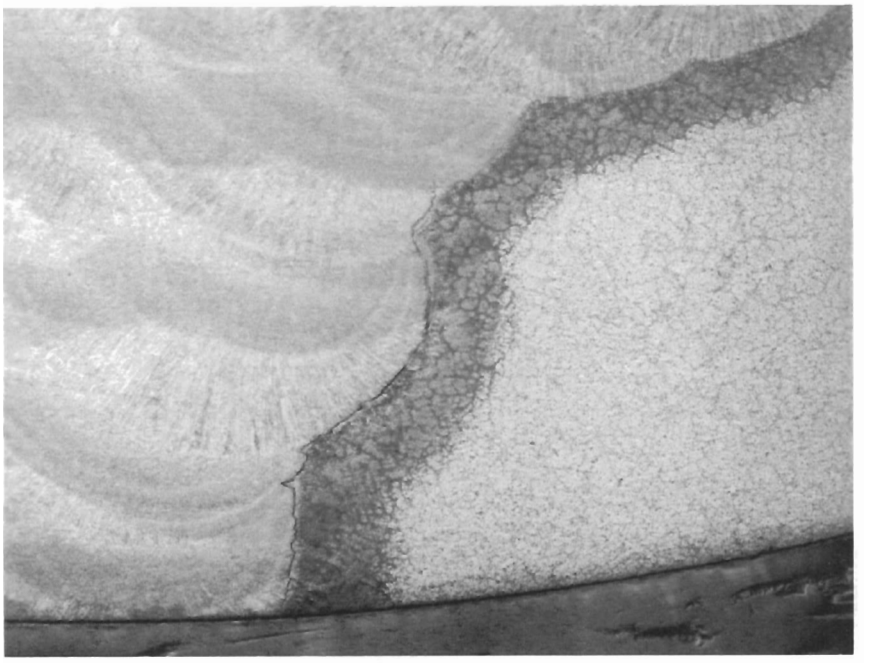

Fig. 13. Failure at Base Metal-Weld Metal Interface Caused by a Metallurgical Notch. (Magnification, 5X)

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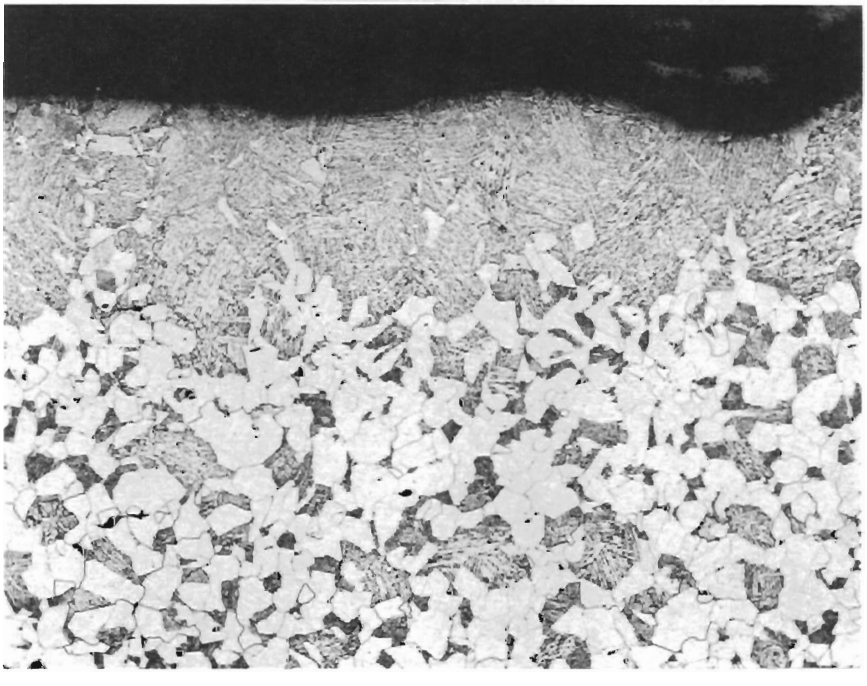

Fig. 14. Surface Carburization in a Seamless Elbow of 1.25% Chromium-0.5% Molybdenum Steel. (Magnification, 100X)

were deposited with a preheat of 500°F and stress relieved at 1325°F, as is normally done, nevertheless failed in the bend test after bending as little as 45 deg, Fig. 15. Where a lug or guide plate for hanger attachment is welded to the outside of an elbow, or where a nozzle connection is made, failure can result during service because of thermal or mechanical fatigue or static stress concentrations of excessive magnitude.

Failures have been the result of hot shortness causing the grain boundaries of the metal to be more liable to rupture at elevated temperatures. Seamless wrought pipe, as well as cast and forged pipe, have failed because of hot shortness.

Final Fabrication and Welding Failures

Fabrication of piping involves (1) forming by hot or cold bending, extruding, or swaging, (2) cutting, (3) welding, and (4) heat treating. When performed improperly, any of the above operations may result in defects which, in high-temperature high-pressure applications, can be and have been responsible for service failures.

Failures resulting from weld defects are by far the most common. The continuously sharpening competitive pattern that places excessive emphasis on cost cutting contributes significantly to defective weldments and produces careless workmanship. It also often reduces the efforts involved in the preparation of detailed welding procedure specifications, the proper training and qualification of welders, and the care taken in inspection.[2]

Pipe butt welds, of course, are usually welded from the outside only. The first weld pass in the root of the pipe joint is thus particularly

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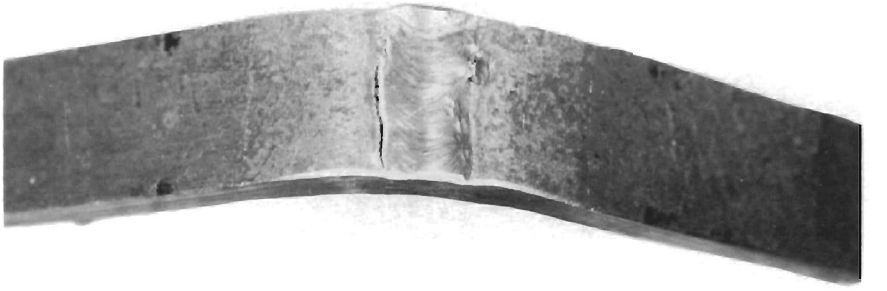

Fig. 15. Cracked Bend-Test Specimen of Butt Weld in Surface-Carburized Elbow.

critical. It is more difficult to make pipe welds than the average plate or pressure vessel weld, which normally can be made from both the inside and outside. Thus, in pipe welds it is very important to avoid defects in the first or root pass.

In high-temperature high-pressure piping systems the first pass is normally made by shielded metal-arc welding against backing rings or by inert-gas tungsten-arc welding. When the latter welding process is used, backing rings generally are not employed. The same or other welding processes may then be used for the subsequent welding passes.

Regardless of the welding process, the use of good joint fit-up is very important. A service failure in a badly fitted pipe joint is illustrated in Fig. 16. The notches formed by the poor fit-up provided conditions for stress-corrosion cracking.

Where the root pass is made without backing rings by the inert-gas tungsten-arc welding process, cracking may occur more readily than in backing-ring welds made by shielded metal-arc welding where the initial weld pass provides more metal of a favorable composition and greater thickness, both of which contribute to a lesser susceptibility to cracking.

It is often not recognized that even though a weld joint has been inspected by various techniques, such inspection is not a guarantee of soundness. Small cracks, inherent brittleness, or other defects may have been introduced by the particular welding process or filler material used.

The difficulty in determining root-pass cracks by x-ray radiography is illustrated in Fig. 17. The root pass in the 6-in. 1.25% chromium–0.5% molybdenum sched.-80 (0.432-in. wall) pipe was welded by the inert-gas

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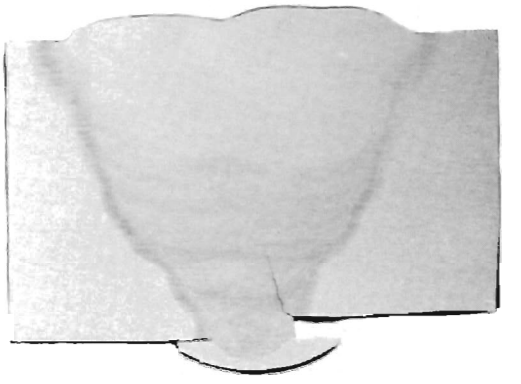

Fig. 16. Root Crack at Notch Formed by Poor Fit-Up Resulting from Normal Pipe Tolerances.

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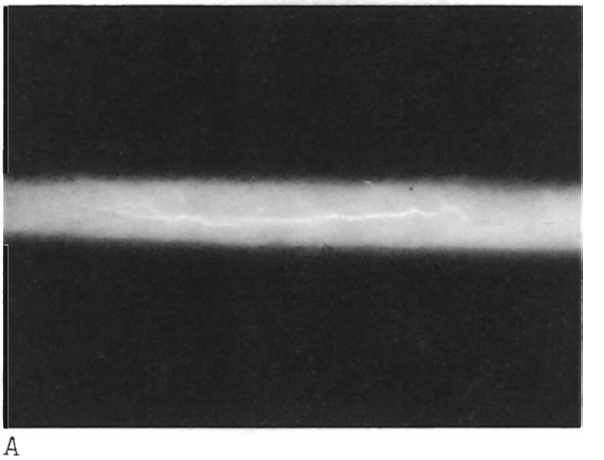
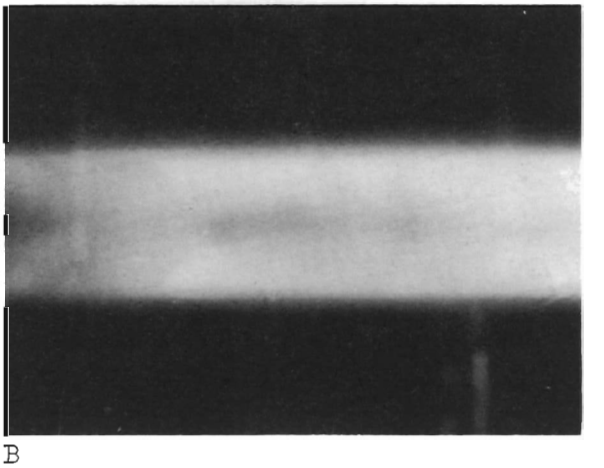


A



B

Fig. 17. Crack in Root Pass of Pipe Butt Weld Made by Inert-Gas Tungsten-Arc Welding. (A) Radiograph of root pass containing a 2.5-in.-long center-line crack. (B) Crack no longer visible in the radiograph after second metal-arc pass with covered electrode.

tungsten-arc process and contained the 2.5-in.-long centerline crack shown in Fig. 17A. As is normally practiced, the subsequent weld passes were made by shielded metal-arc welding with 1.25% chromium-0.5% molybdenum-covered welding electrodes. After the second cover, or fill, pass with the covered electrodes, the root-pass crack was no longer visible by x-ray radiography (Fig. 17B). Obviously the completed weld also appeared sound and without cracks on the radiographic files. Subsequent sectioning of the completed weld, which involved approximately ten shielded metal-arc welding passes, revealed, nevertheless, that the root pass crack was still present. (Fig. 18).

The "disappearance" of the crack was caused by the shrinkage of the first covered-electrode pass deposited over the inert-gas tungsten-arc root pass. The shrinkage drew the crack in the root pass tightly together. Certainly the potential danger of such a crack is not lessened by its being closed mechanically.

Other weld defects in the first pass, such as lack of penetration, slag, or other conditions which cause notch effects on the inner surfaces of the pipe joints, have also resulted in weld failures. Even arc strikes on pipe surfaces can provide conditions for cracking during service, particularly where the service involves thermal or mechanical fatigue and the arc strike produces a highly hardened zone (Fig. 19). Heat treatments that reduce the base-metal hardness adjacent to the arc strike may reduce and eliminate the tendency for this type of cracking.

Localized Heating. Localized torch heating, as in flame cutting, may sufficiently disturb the metallurgical structure to permit corrosion. A failure in a boiler feed-water pipe section, where the area of the steel in which corrosion occurred apparently had been heated to about

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Fig. 18. Cross Section of Weld Illustrating Crack in Root Pass Not Visible by Radiographic Examinations After Subsequent Weld Passes.

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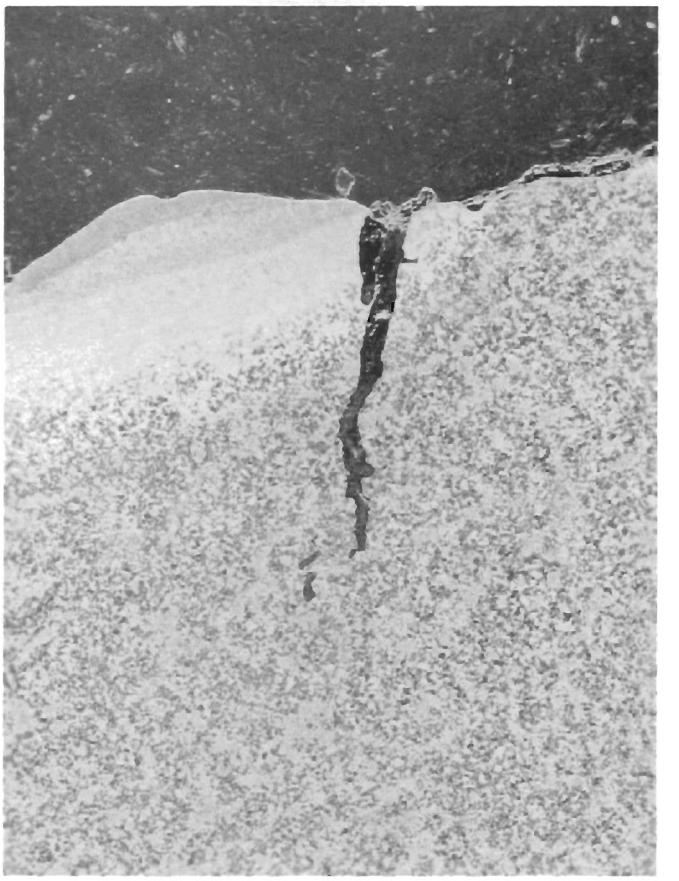

Fig. 19. Crack Starting on Pipe Surface Adjacent to Arc Strike. (Magnification, 18X)

2200°F, is shown in Fig. 20. The heating pattern from the outside of the pipe is evident in the cross section, which was etched to reveal changes in the metallurgical structure.

Uneven localized heating and the resulting hardness differences have also caused a number of service failures in high-temperature high-pressure piping systems. Even though uniform and carefully controlled heating is often extremely important, the industry is becoming, on the whole, less concerned with properly controlled heat treatment.

Weld Stress Relief. The requirements of the ASME Boiler and Pressure Vessel Code and the ASA Code for Pressure Piping have gradually been reduced with respect to post-weld stress-relief heat treatments. Under some conditions post-weld heat treatments may be eliminated completely. For example, for the chromium-molybdenum steels with a total alloy content to 3%, which are used extensively in high-temperature applications, post-weld heat treatments are no longer required where the pipe diameter is 4 in. or less and the wall thickness is less than 1/2 in.

Actually, heat treatments should not be indiscriminately eliminated. Failures have occurred in boiler tubing that had been cold bent. After cold bending, lugs were welded onto the outside. The tubing was then cleaned by pickling and, finally, erected into the boiler. The failures that occurred were related to stress-corrosion cracking that started in the cold bent area from the inside of the tubing underneath where the lugs had been welded (Fig. 21).

Excessively Severe Service Conditions

Failures may occur when the service becomes more severe than the conditions for which the piping was originally designed. Thermal or

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Fig. 20. Failure in Area of Pipe Wall Locally Heated to Approximately 2200°F.

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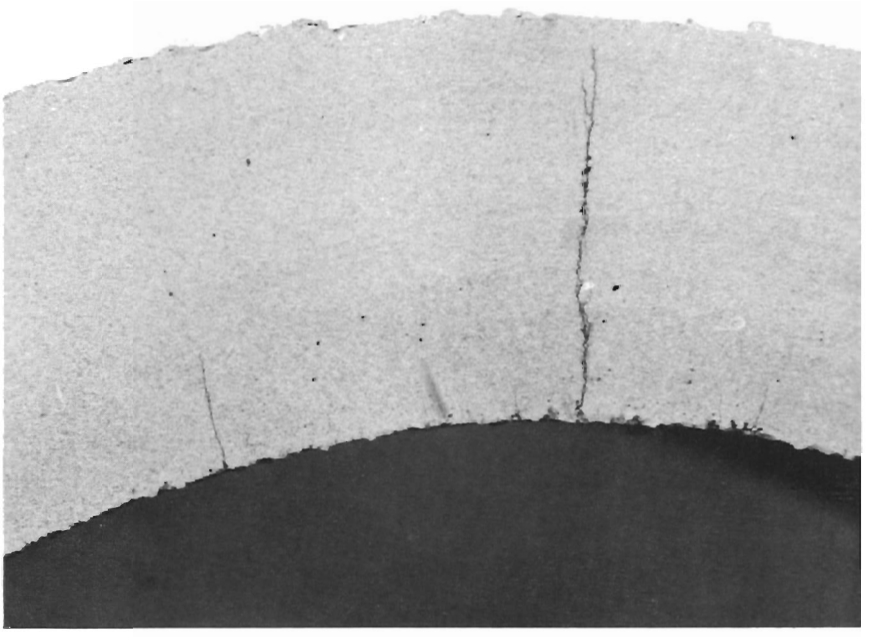

Fig. 21. Stress Corrosion Cracking in Cold Bent Boiler Tubing with Lugs Welded on the Outside. (Magnification, 8X)

mechanical fatigue are usually the most common causes of failures in high-temperature piping systems. Severe localized mechanical stresses have caused or contributed to failures; for example, when the free movement of the piping system is restrained, binding, seizing, or bending may result. Failures have been caused also by overheating, hydrogen embrittlement, creep, and still other service conditions considered excessively severe for the design application or for the materials involved. At times, several of these conditions combine to produce the failure. For example, oxidation inside cracks will resist the normal thermal contraction when the metal cools and may contribute to crack propagation.

Thermal Fatigue. For best metallurgical conditions the temperature of the high-temperature piping systems should be maintained continuously and uniformly. Operation economics often make it advisable, however, to shut down certain plant operations during specific periods. For example, steam power plants often have lowered demand requirements at night and, particularly, on weekends. The reduced load requirements necessitate the shutdown of some boilers at night and of a larger number of boilers on weekends. For example, during a weekend shutdown the piping carrying the steam from a boiler to the turbine may cool from the 900 to 1100°F normal operating temperatures to about 500 to 600°F. This causes contraction of the piping system, the boiler tubing, welded attachments, etc., in accordance with the coefficients of expansion of the materials involved. Sunday night, when the power plant unit is returned to normal operations, the boiler tubing, steam piping, and turbine casings are heated up again by the steam produced in the boiler. Thermal expansion occurs in all metal components, and the piping systems expand to their normal position. Over

a period of months, the resulting thermal fatigue may be more harmful to the materials involved, particularly at mechanical or metallurgical notches, than continuous operation would be at a particular temperature over a period of many years.

Cracking in a main steam line caused by thermal fatigue is shown in Fig. 22. Sometimes single cracks occur; however, more often a whole family of parallel cracks occur that usually start from the inside of the pipe. The inner surface is subject to more severe thermal shock than the outer surface. These parallel cracks, Fig. 23, gradually propagate until they reach the outer surface.

Many failures have occurred in pressure-reducing sections in which water is sprayed on the pipe inside. The quenching effect results in temperature cycling over the range 300 to 500°F and causes failures across the pipe wall, as in Fig. 24. The insertion of stainless steel liner plates inside of the pipe generally does not provide any improvement, since these liner plates fail also, as in Fig. 25, and usually just as readily.

Mechanical Fatigue. Even under continuous high-temperature operation, some failures have occurred (Fig. 26). Usually these can be related to some degree of mechanical fatigue resulting from pipe movement, vibrations, restraints preventing free movement, or other conditions. Such failures often show a number of parallel cracks (Fig. 27).

Overheating. A failure caused by overheating of boiler tubing is illustrated in Fig. 28. Such failures are usually confirmed by considerable differences in the microstructure (Fig. 29).

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Crack →

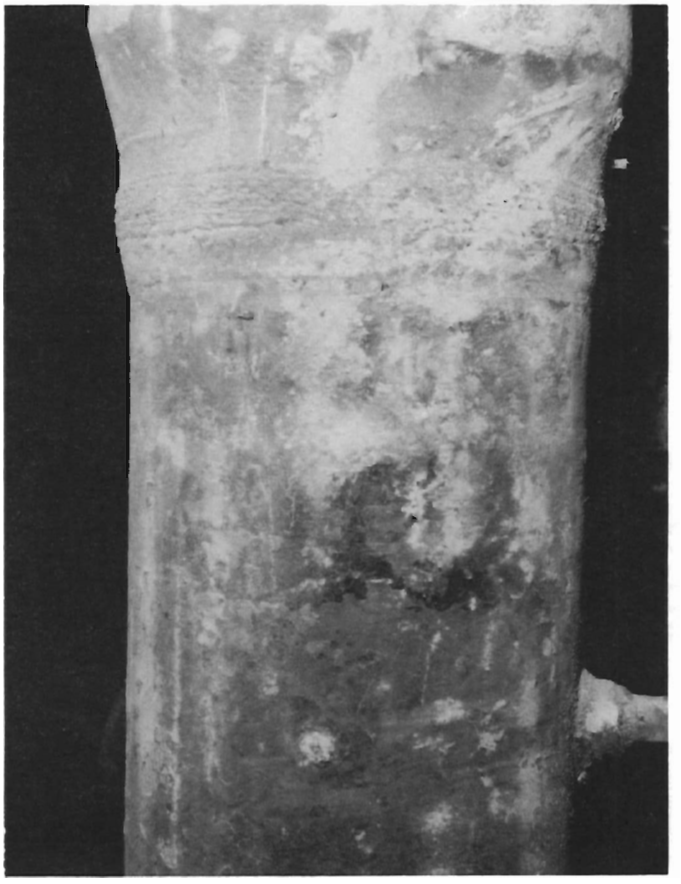

Fig. 22. Cracking in Main Steam Pipe Section Caused by Thermal Fatigue.

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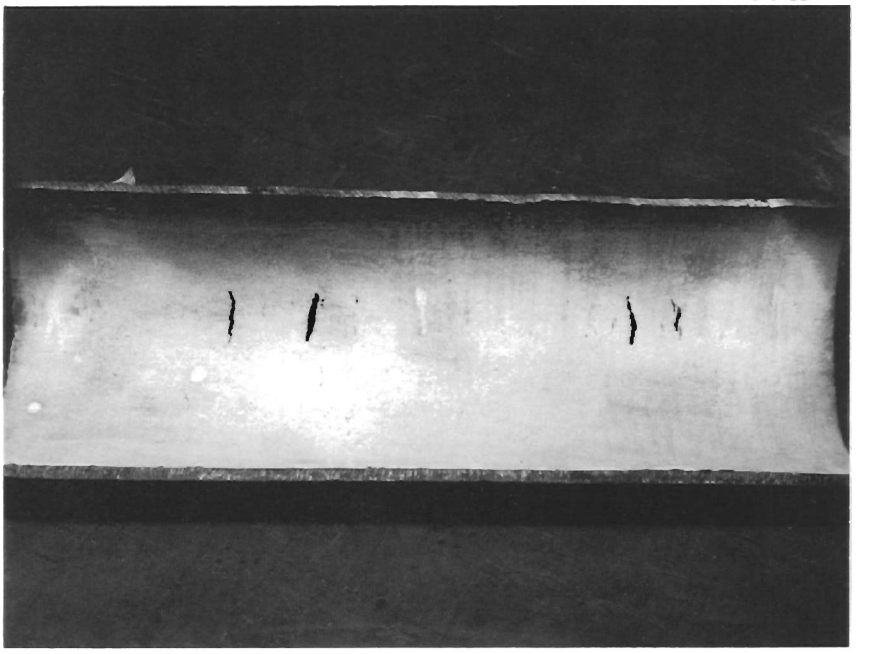

Fig. 23. Parallel Cracks on Inner Surface of Pipe Section Subject to Thermal Fatigue.

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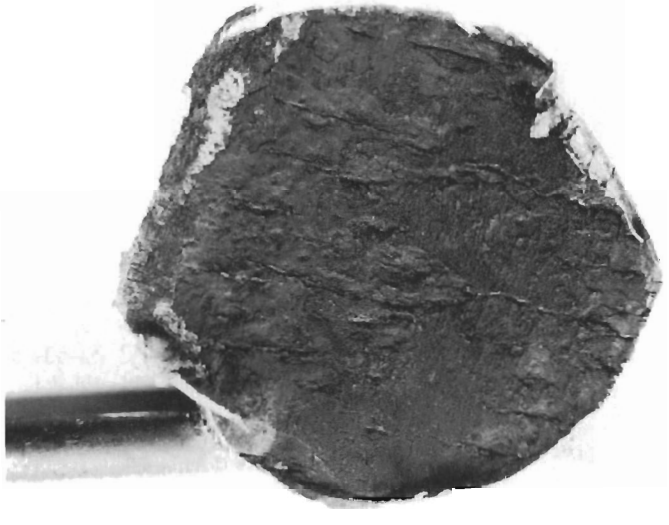
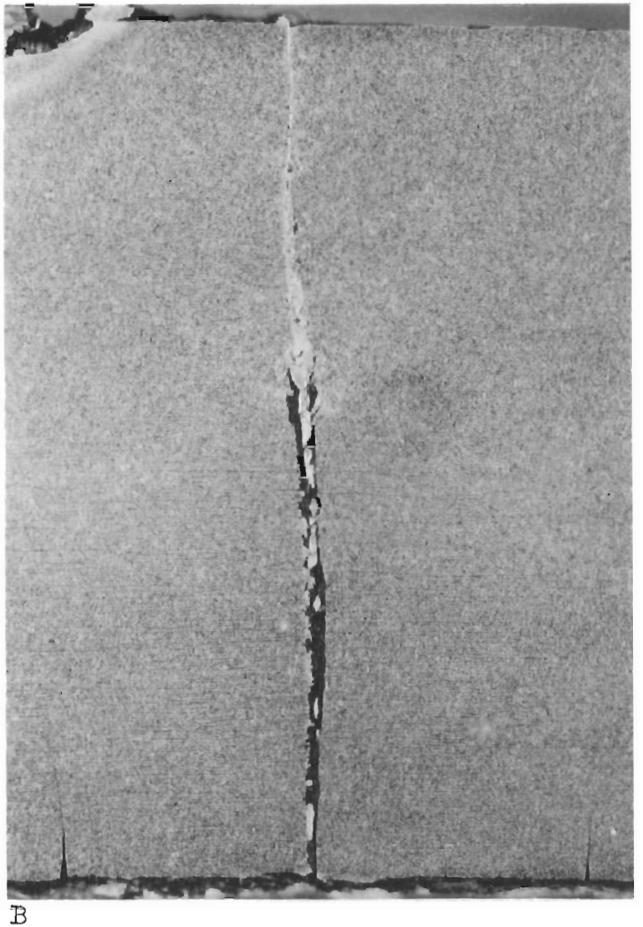


A



B

Fig. 24. Cracks Caused by Thermal Fatigue in Pressure-Reducing Section. (A) Photograph of inner pipe surface. (B) Cross section magnified 5X.

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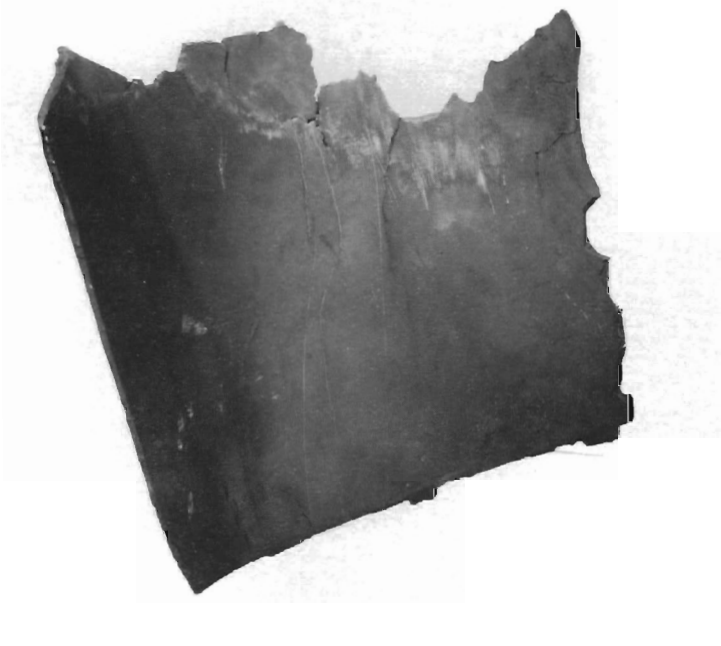

Fig. 25. Cracks in Stainless Steel Liner Plate Welded on Inside of Pressure-Reducing Pipe Section.

Unclassified
Photo 59642

Fig. 26. Cracks in Pipe Section Experiencing Some Mechanical Fatigue. (A) Photograph of Surface. (B) Print of radiograph.

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Photo 59643

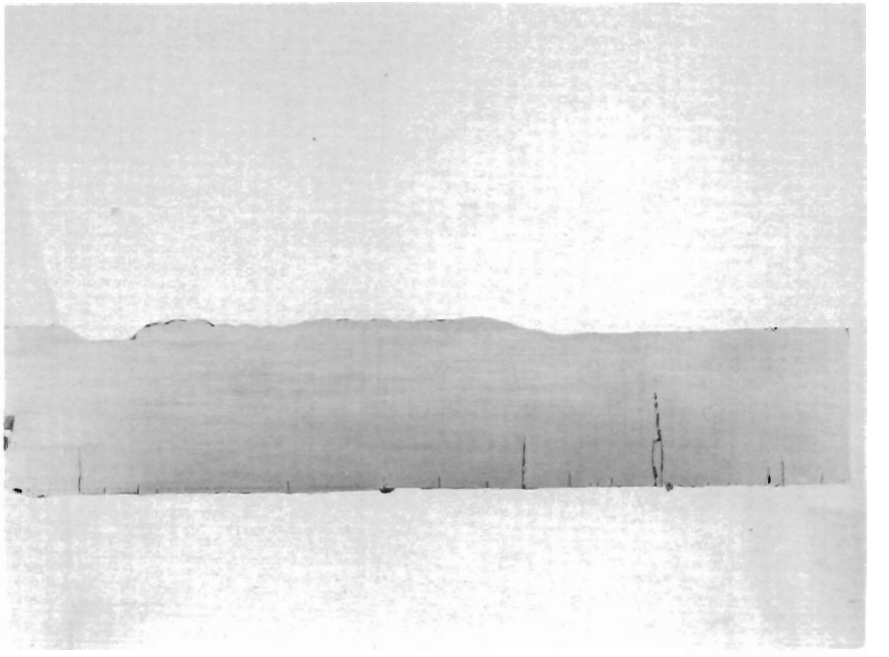

Fig. 27. Family of Parallel Cracks Resulting from Mechanical Fatigue.

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Photo 59644

Fig. 28. Failure in Boiler Tube as Result of Overheating.

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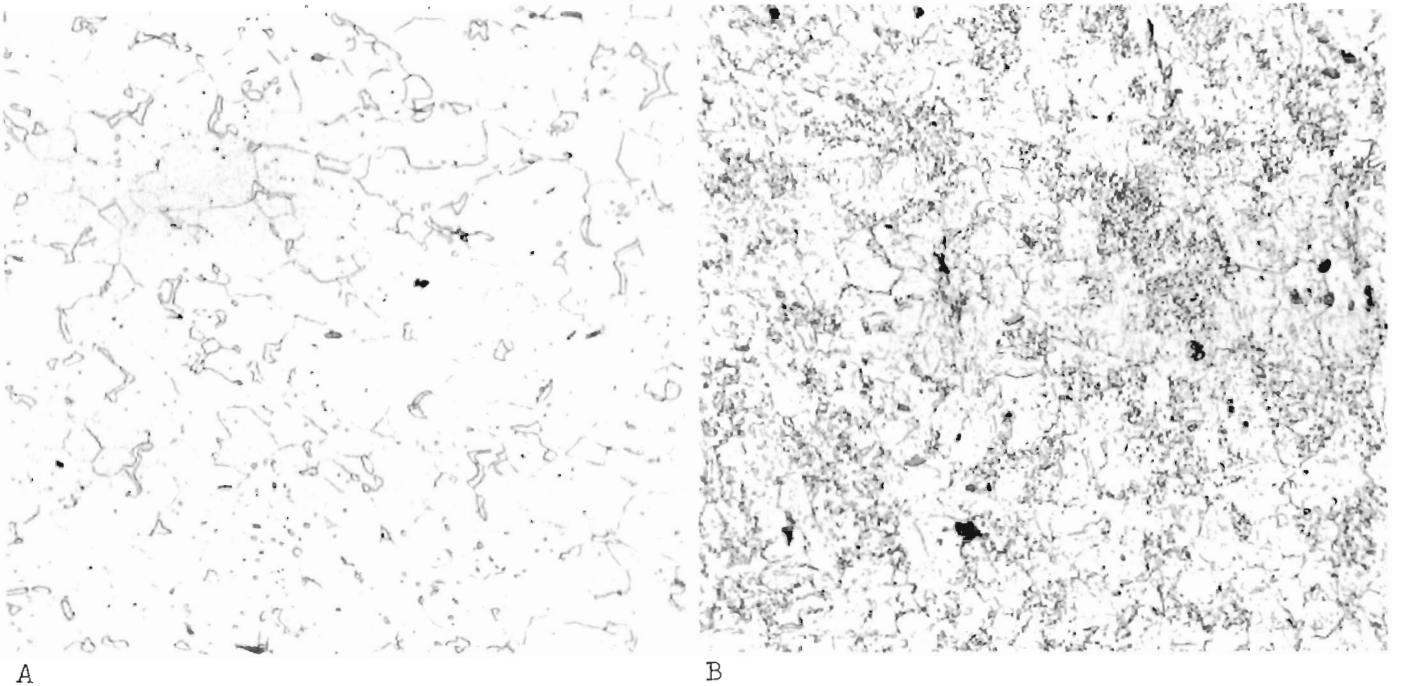
Photo 59645



A B

Fig. 29. Photomicrographs Showing Difference in Microstructure Caused by Overheating that Resulted in Service Failure. (A) Fire Side. (B) Opposite side. (Magnification, 100X)

Hydrogen Embrittlement. Hydrogen embrittlement is shown in Fig. 30. Such failures involve service at temperatures between 600 and 1000°F and are usually preceded by considerable localized scaling (Fig. 31). Moreover, the microstructure shows decarburization (Fig. 32).

Copper deposits on the inside of high-temperature piping can also contribute to high-temperature failures. It must be recognized also that some failures in high-temperature piping may actually involve initiation or propagation of cracks during shutdown periods when the piping is at low temperatures. Static stresses are likely to become increasingly severe as the piping system cools.

Incorrect Material Identification

Incorrect identification of materials for critical piping service applications has been a far more frequent cause of failure than might be expected. The error may have been made by the fabricator or by the erector handling the materials subsequently or by the user replacing piping components. Changing the service conditions in the operating plant to more severe conditions and assuming that some materials are of a different or more highly alloyed grade have also led to failures.

The danger of mix-up is particularly great when the respective identification markings have not been applied directly to the surface of the material by die stamping or, on some materials, by electric etching. Nevertheless, a significant number of errors in identification have occurred even on materials marked by die stamping.

Two type 304 stainless steel couplings torn loose from a type 304 stainless steel vessel are shown in Fig. 33. To avoid possible mix-up

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Photo 59646

Fig. 30. Failure in Boiler Tube as Result of Hydrogen Embrittlement. (Reduced to 1/2 size)

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Photo 59647

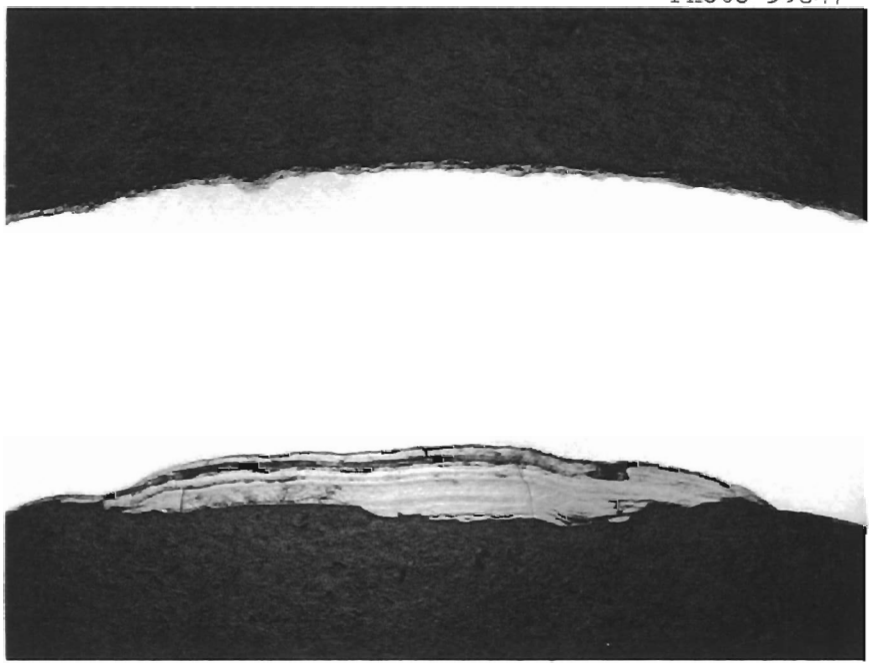

Fig. 31. Cross Section of Boiler Tube Showing Localized Scaling that Preceded Hydrogen Embrittlement. (Magnification, 5X)

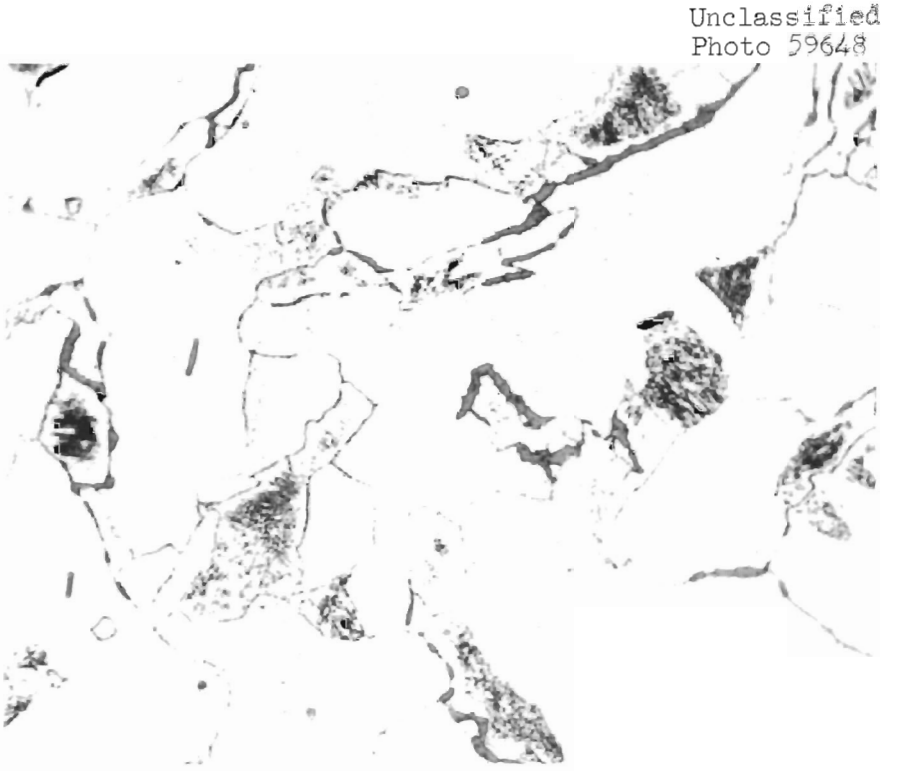

Fig. 32. Photomicrograph Showing Evidence of Decarburization in Boiler Tube That Failed Because of Hydrogen Embrittlement. (Magnification, 1000X)

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Photo 59649

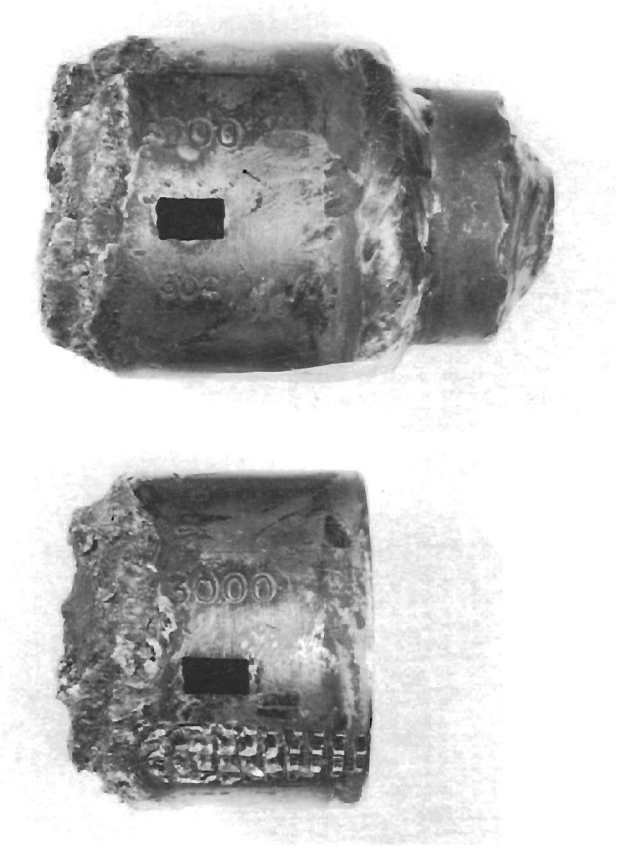

Fig. 33. Failure Along Weld of Two 3000-lb Couplings Identified as Type 304 Stainless Steel, Which Was Actually Monel.

in the fabricator's shop, the couplings had the type 304 markings and 3000-lb pressure rating applied directly to the surface by the manufacturer. Certified mill test reports also attested that the chemical composition conformed to the requirements of the American Society for Testing Materials. The type 304 couplings were welded to the type 304 vessel and filled in a service test. Subsequent examination revealed that the couplings actually were Monel alloy 400 (70% Ni–30% Cu). Welding of Monel alloy 400 with stainless steel electrodes results in brittle joints (Fig. 34) because of the detrimental effect of copper in stainless steel weld metal.

An example of a high-temperature failure in an incorrectly identified boiler tube is shown in Fig. 35. The material should have been 2.25% chromium–1% molybdenum, but actually it was carbon steel.

Responsibility of Specifications

On the whole, service failures in high-temperature high-pressure piping and tubing systems are on the increase. Although a number of factors are responsible, one involves a decreasing concern among purchasers with respect to the quality of the pipe materials and fabrication they purchase. Where low price has become the only or primary consideration in the evaluation of quotations, the material or fabrication obtained may not meet the requirements of applicable codes and service, and may not even represent good workmanship. Excessive low-price emphasis by purchasers has forced some mills and suppliers to reduce supervision, inspection, and quality control. It has led some contractors and inspection agencies to employ short cuts or even deception.

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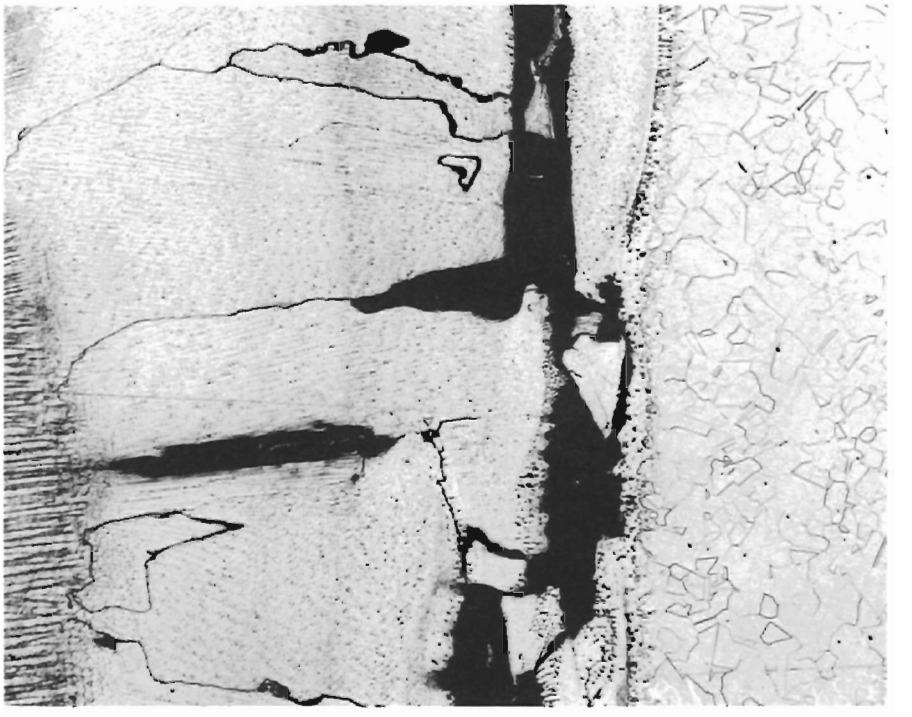

Fig. 34. Photomicrograph of Interface Between Type 308 Weld Deposit and Monel (70% Ni–30% Cu) Base Metal Illustrating "Hot Short" Cracking Caused by Copper Penetration. (Magnification, 100X)

Unclassified
Photo 59651

Fig. 35. Failure in Boiler Tube Resulting from Incorrectly Identified Tube Material. The material was carbon steel but should have been 2.25% chromium–1% molybdenum steel.

Specifications are often not realistically prepared in sufficient detail to ensure sound materials and fabrication. Moreover, welding specifications are frequently prepared by persons who do not have intimate knowledge of welding, its capabilities, and its limitations.

Inspection requirements may not be based on realistic considerations. Too frequently it is not recognized that each specific destructive or non-destructive test evaluates only a particular material or weld in comparison with another material tested under the same conditions. The interpretation of test results and their correlation with the behavior of the material under actual service conditions is at best only approximate. Quite frequently, the results have little or no relation to the actual service performance of the piping material that is to be welded.

Perfection Must Be Tempered by Realism

The author definitely does not intend to imply that much stricter specifications covering materials, fabrication, welding, and inspection are needed. Instead more reasoned intelligence needs to be applied to specifications. A number of excessively tight special specifications already exist that have requirements of perfection so unrealistic that they have no relation to the actual service requirements of the piping components. In fact, unrealistic and excessively severe quality requirements may actually cause defective or less serviceable piping components. This may be brought about in either or both of two ways. First, a reliable fabricator who understands inspection by employing all of the fabrication and inspection know-how at his disposal may find that he cannot produce the component at a "reasonable" or competitive cost. An inexperienced

fabricator, who does not inspect every step in the fabrication of the component with the same detail, may get the job and miss detecting faults that later on contribute to a service failure. Second, repeated repairs of minor weld defects that show on radiographic films quite often cause other joint defects, notably metallurgical notches or tight root-edge creasing, which are not normally detectable by present inspection techniques but are likely to be more detrimental than the minor internal faults.

Good engineering practice must be based on realism and must require a level of quality of materials, fabrication, and welding that meets the demand of the particular service. Too many engineers who are not familiar with materials, fabrication, welding, and inspection write requirements into specifications that involve extremely costly materials and procedures but which do not improve the reliability or the life expectancy of the product. By concerning themselves with unimportant (and usually costly) requirements, important considerations involving welding, heat treating, or inspection procedures are often confused or neglected. One requirement written into a number of specifications states, for example, that tungsten inclusions in stainless steel weld deposits are not permissible. Since very small tungsten inclusions from the welding arc can readily "drop" into the molten weld puddle and be contained within the weld, careful radiographic inspection is necessary to detect these. Grinding out and rewelding all weld areas containing tungsten inclusions may require repair of 25% or more of the welds in a piping system. This will remove the fine tungsten inclusions, and there is no actual evidence that they have ever caused service failures. The rewelding may result in more serious defects, however, such as very fine fissures and cracks not detectable by

the inspection techniques normally applied. It has been claimed that the very small tungsten inclusions represent hard particles within stainless steel and that because of their different thermal coefficient of expansion they can start microfissures during service involving thermal or mechanical fatigue. On the other hand, stainless steel, and in fact all alloys, contain carbide particles and other fine inclusions that differ substantially in hardness and coefficients of expansion from the main body (matrix) material. These, like tungsten, should also not be permitted. By logic, this would disqualify all stainless steels, all other steels and, in fact, all materials from consideration. Actually, tungsten inclusions should be considered on the same basis as porosity or slag inclusions at the levels permissible in weld deposits.

Adverse experiences with certain specific fabrication, welding, or inspection procedures are frequently used as the basis of specification requirements written for the next job without a careful evaluation of causes for the adverse results. For example, adverse experience with consumable insert rings used in pipe welding on one project led to their being condemmed by the same group on other projects. It was not recognized that the problems might have arisen from consumable inserts of unfavorable cross section or the improper use of these inserts. In fact, consumable welding insert rings manufactured from welding wire compositions have been used on many projects where they have produced pipe welds consistently of the highest uniform quality. Many other examples could be cited of unrealistic fabrication requirements.

There is a need also for learning more about causes and conditions leading to service failures. Various nonhomogeneities in materials,

mechanical and metallurgical notches, and weld defects should be evaluated in realistic tests simulating actual service conditions. Many types of nonhomogeneities rarely, if ever, cause or contribute to service failures. Requiring their removal or greater material perfection might increase product costs substantially and might double or triple welding costs. Furthermore, important projects might be delayed needlessly, and additional cost would result.

Inspection also needs to be tempered with realism. Millions of dollars worth of good welds have been cut out or repaired because inspectors did not know how to interpret nondestructive test results. Often inspectors from one qualified inspection agency may pass a weld, only to have it rejected by another agency. Resolving the ensuing arguments may delay urgent projects and increase the cost needlessly to a point where the inspection costs often far exceed the material or welding costs. The responsible fabricators may then lose interest in the most important high-quality programs and leave the field to less qualified contractors.

Acknowledgement

The failures described in this article represent a few examples from investigations conducted by the Industrial Piping Laboratory on a consulting basis for various public utilities, nuclear power plants, refineries, chemical plants, paper mills, and other commercial plants.

References

1. H. Thielsch, "Thermal Fatigue and Thermal Shock," Welding Research Council Bulletin No. 10, April 1952.

2. H. Thielsch, "What an Engineer Must Know about Codes, Specifications, and Inspection for the Prevention of Pipe Failures," Heating, Piping Air Conditioning, 34(8): 98–103 (August 1962).

I. GENERAL SAFETY CRITERIA

AEROSPACE NUCLEAR SAFETY PROGRAM

W. K. Kern* G. B. Conner**

Operation of nuclear reactors and radioisotopic thermoelectric generators in space presents potential nuclear safety problems that are unique when compared with the nuclear safety problems of terrestrial reactors. The Atomic Energy Commission's statutory responsibilities, as assigned by the Atomic Energy Act of 1954, as amended, to provide for the protection of public health and safety with respect to the applications of nuclear energy have assumed an entirely new aspect. This was recognized early, and an Assistant Director for Nuclear Safety was established within the Division of Reactor Development to help fulfill these responsibilities.

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**Capt. Gerald B. Connor, a graduate nuclear engineer, is presently assigned to the U. S. Atomic Energy Commission as the Aerospace Nuclear Safety Analyst for the Division of Reactor Development. In this assignment, he has been responsible for nuclear safety analysis and evaluation of the aerospace nuclear safety programs and proposed aerospace flight plans and operations. The aerospace projects involved include the SNAP reactors, SNAP radioisotopes, and Pluto. Prior to this assignment, Capt. Connor served with the Aircraft Nuclear Propulsion Office in the U. S. Atomic Energy Commission in the Aircraft Reactor Safety Branch. His professional career also includes three-year tours as a special weapons nuclear supervisor and as a SAC pilot. Capt. Connor has been actively engaged in all fields of aerospace nuclear safety and is presently participating in the preparation of an aerospace nuclear safety handbook.

In nuclear safety, the basic study, laboratory investigation, and analytical approaches to the problems are essential but are not believed to be sufficient. The final "conviction" or demonstration that nuclear accident phenomena are understood and that measures have been taken to adequately "contain" the consequences or to reliably prevent an accident must come from full-scale proof testing of the integrated system. This philosophy is especially pertinent to aerospace systems, since the systems must operate within strange and rigorous environments not heretofore encountered.

The safety of the AEC's aerospace SNAP (Systems for Nuclear Auxiliary Power) devices and primary nuclear power or propulsion systems must be assured before they can be used for space applications. The four periods in the operational life of an aerospace nuclear system that require specific safety considerations are: (1) prior to and during lift-off (prelaunch operations), (2) ascent, (3) orbital injection, and (4) re-entry following mission completion. Some aspects of the safety of various aerospace vehicles were discussed in two previous Nuclear Safety articles.[1,2]

Nuclear systems such as the various SNAP reactors will have essentially no fission-product inventory during launch and orbit-injection operations, and thus the primary problem will be to prevent criticality and possible nuclear excursions in the event of an abort. For isotope-powered SNAP units, a diametrically opposite problem confronts safety personnel, since the generator contains its greatest nuclear strength at launch. Thus each system must be approached as a separate entity in the demonstration of the safety designed and engineered into it. Until

capability for recovery, boost into solar orbit, or other means can be assured to prevent the dispersal of radionuclides within the biosphere, re-entry burnup is the mechanism chosen to prevent contribution to either local or world-wide radiation levels.

Scope of Aerospace Nuclear Safety Testing Program

The first step in establishing a meaningful aerospace nuclear safety program required recognition of all the unique aerospace environments, both normal and abnormal, to which the nuclear system could be subjected. The nuclear system was studied in all its various possible environments from assembly to re-entry upon mission completion, and all credible accidents were cataloged. A typical, but simplified, example derived from this procedure is outlined below:

1. The incidents that can occur during transport to the launch base are:
    a. Road collision, overturning, or fire.
    b. Road accident followed by water reflection.
2. Ground-handling procedures can lead to:
    a. Fire, explosion, crane fall, crushing, etc.
    b. Criticality as a result of procedural errors.
3. Prelaunch checkout can pose safety problems because of:
    a. Crane failure in which the reactor falls 100 ft.
    b. Procedural errors that activate the startup procedure.
    c. Vehicle collapse as a result of tank pressurization loss.
4. Vehicle fueling and launch can cause:
    a. Accidental actuation of startup radio command.

b. Accident at launch pad during fueling or liftoff.

c. Inflight accidents that would result in land impact.

d. Flight accidents that would result in sea impact.

5. Orbital nuclear-electric operation may perturb the system but, because a satellite's orbital lifetime is fixed by natural laws, the orbit constitutes an ideally safe reactor-operating site. While the system is in space, no accident or malfunction can result in hazards to persons.

6. Orbital decay and re-entry may pose additional difficulties, including:

a. Re-entry after failure of primary reactor shutdown.

b. Re-entry after failure of all shutdown systems.

Once the possible incidents are postulated, they can be grouped into families of similar problems, and, further, the postulated maximum credible accident can be identified. Then, in logical sequence, the problems the normal system and its components will encounter can be established. Based on the solutions to these problems and an added safety factor, the system can be engineered and tested. Actually, a continuing safety analysis and assessment of both the nuclear unit and the complete system must be performed as development progresses.

Although the prime contractor is responsible for safety, it became obvious that the aerospace nuclear safety testing program was too large and complex a task to be supervised adequately by the individual program offices of the AEC's Division of Research and Development. The experiments to be conducted and the environments to be studied were closely related and applicable to many other aerospace nuclear projects.

Therefore, a large portion of the aerospace nuclear safety testing program was assigned to the Assistant Director for Nuclear Safety. This office, working closely with program offices, studies the environments and performs the necessary full-scale field tests for all aerospace nuclear projects. In fulfilling this mission, this office is in a position to independently assess the safety factors and features of SNAP and primary nuclear propulsion systems. The aerospace nuclear safety program consists of four major portions: (1) research and development, (2) ground environment studies, (3) flight testing, and (4) reactor kinetics.

Research and Development

The research and development phase of the nuclear safety program is concerned with the basic physics of the upper atmosphere, the thermodynamic heating rate, nuclear component and material ablation mechanisms (such as diffusion and dispersion), and resident times of ablated nuclear system particles. Additionally, this portion of the program involves the independent safety assessment of the nuclear system, since knowledge of normal and abort environmental parameters is established through analysis of the possible modes of accidents, as discussed previously. Several research flights to investigate upper atmospheric phenomena as they pertain to nuclear systems are to be conducted in the near future. The toxicity of nonradionuclide materials and vacuum welding will also be studied. It is hoped that design and engineering concepts and techniques will evolve from these "across-the-board" research projects that will aid in future nuclear systems development.

Ground Environment Studies

The ground environment studies encompass experiments that simulate ground or launch-pad aborts and normal and abnormal operational environments from liftoff to orbital injection. In various experiments the nuclear system and its components are being subjected to the rigorous environmental conditions that can exist in case of a missile accident. Components and complete systems are impacted against granite, soil, mud, and water at selected velocities simulating the various conditions to which the nuclear device may be subjected. For instance, a fall from the top of the booster vehicle would result in a rather low-speed impact, while a return to earth because of failure to inject into orbit or a late abort would result in a high terminal velocity, the ultimate speed being dictated by the physical parameters of the system. Thus impact tests are run at speeds as low as 100 ft/sec and as high as 800 ft/sec. Under these conditions, SNAP reactors should rapidly disintegrate and spill the core fuel rods in such a way that criticality would be improbable. Isotope-thermoelectric units, on the other hand, must contain the fuel; thus the isotope container must be designed to maintain integrity.

In a launch-pad abort, not only does the nuclear system have to withstand impact forces, it must also withstand the extreme temperature environment that exists upon propellant ignition. Experiments are being conducted using simulated missile fuel conflagrations in which the nuclear components are immersed. The temperature effects on the components are being investigated and evaluated. The possible dissemination of respirable toxic materials in the SNAP or nuclear propulsion system, such as beryllium, is being observed. Since an explosion caused by missile

fuel conflagration could subject the nuclear device to abnormal impulses and overpressures, explosion tests are also being conducted.

Chemical interactions that could result from launch-pad aborts are being evaluated. The reactions of liquid propellants or hydrogenous liquids with reactor coolants (for example, NaK) and other materials are being studied to determine their extent. Electromagnetic radiation and vacuum welding effects on the reactor and its controls are being determined and evaluated.

Flight Test

The flight-test phase of the nuclear safety program for aerospace systems involves flight demonstrations of the model or prototype system. The safety proof of the system lies in this type of experiment. Orbital decay and abort trajectories of various configurations will be flown at conditions as close as experimentally feasible to the actual operating conditions. Acquisition of the maximum amount of meaningful data with respect to recovery capabilities will be the secondary objective of each flight, the primary objective being the nuclear safety demonstration of the system.

A vigorous aerospace nuclear safety program of many facets has been initiated on an accelerated basis to ensure empirical verification of analytical safety assurances on a timely basis. One of the first experiments to be conducted was the evaluation of the mathematical re-entry model upon which isotopic thermoelectric generator burnup upon re-entry from orbital mission was based. The experiment consisted of encapsulation of pyrophoric flare materials (alkalis) into cannisters of various

shapes and wall thicknesses. These, in turn, were enclosed in a scientific passenger pod integrated into a developmental Atlas booster vehicle. A parabolic ballistic trajectory was flown with the capsules being ejected from the pod above the sensible atmosphere (>400,000 ft). Spectrographic data were obtained on encapsulation material burnup and pyrophoric flaring of the alkaline materials contained therein upon re-entry into the atmosphere. The analytical model was used to predict the position and altitude of these events, and the results of the pod experiments indicated that the model utilized was accurate within the limits of experimental error. Thus, it can now be confidently stated that the re-entry model yields a true picture of the aerothermodynamic events to which the re-entering system will be subjected.

The first complete USAEC safety demonstration flight is scheduled for the near future. Its primary objective will be to prove the safety design features engineered into an aerospace reactor. The design is such that rapid mechanical disassembly is achieved, making early exposure of the reactor core to the re-entry heat possible. Ejection of the fuel elements into the high-altitude air stream should result in exposing the individual elements to the maximum amount of heat and should provide the best opportunity for complete burnup.

Space Reactor Kinetics

One of the most serious accidents postulated for SNAP reactor systems is a nuclear excurison induced by the accidental immersion of the reactor in water or by the inadvertent rotation of the reflector drums to a critical position. Either of these conditions could conceivably

occur in the event the reactor should fall from the vehicle at the time of launch. Accordingly, the safety testing of SNAP reactor systems will include a series of transient tests, including excursions in the destructive range, to determine the maximum energy release to be expected should an accident occur. The ultimate shutdown mechanism for the excursion will also be determined.

The first destructive excursion test is scheduled for July 1963. Although the probability of this type of accident is small, the possible effects must be investigated because of their serious hazard potential. Since significant data regarding the static and dynamic behavior of SNAP reactor systems will be acquired from these tests, they will also contribute significantly in determining design parameters for future systems.

Conclusions

A comprehensive, vigorous aerospace nuclear safety program has been initiated to demonstrate and assure that the use of SNAP and primary nuclear propulsive power packages will not present a radiological hazard to the world's populace. It is to be noted that, with proper operational procedures, an incident that would release radionuclides into the biosphere is held to be most unlikely, since the nuclear system will be developed and engineered with a high degree of safety integrity. The aerospace nuclear safety program will investigate the maximum credible incident, even though the probability of its occurrence will be very low. Nuclear safety experiments will analyze the safety, integrity, and validity of the nuclear system in the most rigorous environments. Thus,

lesser incidents, which are more likely to occur, will not become of consequence. The results of the independent assessments by the aerospace nuclear safety program of the safety factors involved in the use of an operational aerospace nuclear device will establish a very high degree of safety confidence.

Additional comments on this subject are available in references 3 through 12.

SAFETY IN GAS-COOLED POWER REACTORS

The interest of the United States in gas-cooled power reactors extends, historically, back to the early days of the Manhattan Project, but the detailed engineering development of such systems has evolved only recently. The United Kingdom, through development of its Calder Hall reactors, established a much earlier program of detailed system engineering and now has built or has under construction plants with several thousand megawatts of nuclear-electrical capacity based on the Calder Hall reactor technology. The current U.S. gas-cooled reactor programs were initiated in 1957, and a number of experimental and prototype gas-cooled reactor systems are now in various phases of design, development, and construction in the United States. At the present time, however, only one closed-cycle reactor system, the comparatively small gas-cooled reactor experiment (GCRE) at the National Reactor Testing Station (NRTS), has been operated at significant power levels. Safety requirements are therefore still being studied and are subject to continuing change. In the United States, safety evaluations of reactor designs are carried out primarily on the basis of the water-cooled reactor systems for which the technology is well established. In the United Kingdom, the Calder Hall systems have provided the reference basis, and technological changes are evaluated on the basis of the safety precedents established for these early gas-cooled reactor systems.

The safety reference for the two countries has resulted in somewhat different concern for activity releases through fuel element failure, since the very highest rated gas-cooled reactors under consideration in

the United States are a factor of 5 lower in power density than the water-cooled reactor systems, whereas the power density of the Calder Hall systems is being scaled up in power density by factors of 10 to 30 in large gas-cooled reactor systems. Nevertheless, as the technology advances, the safety approaches of the two countries appear to be converging on a common ground.

Classes of Gas-Cooled Reactor Systems

The gas-cooled reactor systems now under development, see Table I-1, include such a wide variety of engineering differences that a single set of design characteristics cannot be specified. By considering their fuels, moderators, coolants, and power-recovery systems separately, some distinctive features can be established for the several systems.

Reactor Fuels. The various types of reactor fuels being used or considered are described below:

1. Natural-uranium Magnox-clad fuel elements are used for the United Kingdom's Calder Hall type of reactor, and both the United Kingdom and France are building a substantial power complex based on such fuel elements. Their upper temperature is limited[13] by the Magnox melting point of 580°C (1076°F).

2. Enriched UO_2 fuel with metal cladding is used for reactors with high-temperature capability. Reactors that will utilize this type of fuel are the U.S. experimental gas-cooled reactor (EGCR)[14] and the British advanced gas-cooled reactor (AGR),[15] which will have type 304 stainless steel-clad fuel rods. Earlier designs for the AGR also considered beryllium cladding and the recently abandoned Florida West Coast Nuclear

Table I.1 SUMMARY OF OPERATING AND DESIGN DATA OF IMPORTANT GAS-COOLED REACTORS

	EGCR	AGR	HTGR	Dragon Project	Calder Hall, One Reactor	Hinkley Point, One Reactor	FWCNP	Sanderson & Porter Pebble-Bed Reactor	ORNL Pebble-Bed Reactor (PBR)	ORNL Pebble-Bed Reactor Experiment (PBRE)	Brown Boveri and Krupp (AVR)	MGCR	ML-1	UHTREX (Turret)	EBOR	GCRE-1
	Kaiser and Allis Chalmers	UKAEA	General Atomic	UKAEA	UKAEA	English Electric	General Nuclear Engineering	Sanderson & Porter	ORNL	ORNL	Brown Boveri and Krupp	General Atomic	Aerojet General	Los Alamos	General Atomic	Aerojet-General
	Prototype, power	Experimental, power	Prototype, power	Power study	Pu production and power	Power	Protype, power	Power	Power	Experimental	Power	Prototype for ship propulsion	Prototype, power	Experimental	Experimental	Experimental
	Under construction	Under construction	Under construction	Under construction	Operating	Under construction	Conceptual	Conceptual	Conceptual	Conceptual	Under construction	Conceptual	Operating	Design	Construction	Operation completed, being dismantled
	1964	1962	1963	Unknown	1956	1960–1961	Discontinued	Discontinued	Discontinued	Discontinued	1963	Discontinued	1961	Unknown	Unknown	
	Indirect	Indirect	Indirect	Heat dump	Indirect	Indirect	Indirect	Indirect	Indirect	Indirect	Indirect	Direct	Direct	Heat dump	Air heat dump	Water heat dump
	84.3 25.7	100 28	115 40	20.0 None	182 42	980 250	159 50	337 125	800 330	10 3	49 15	74.4 50	3.3 0.33	3 None	10 None	2.2 None
	Helium	CO_2	Helium	Helium	CO_2	CO_2	CO_2	Helium	Helium	Helium	Helium	Helium	N_2 or air	Helium	Helium	N_2–0.5% O_2
	Graphite	Graphite	Graphite	BeO	Graphite	Graphite	D_2O	Graphite	Graphite	Graphite	Graphite	BeO	H_2O	Graphite	BeO	H_2O
	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite	D_2O	Graphite	Graphite	Graphite	Graphite	BeO		Graphite	BeO	Lead
, ft									3	3					~0.6	0.176 to 0.5
	UO_2	UO_2	UC and ThC in graphite	UC and ThC in graphite	U metal	U metal	Cored UO_2	UO_2-ThO_2 in graphite matrix	UO_2-ThO_2 or UC_2-ThC_2 in graphite matrix	UO_2-ThO_2 or UC_2-ThC_2 in graphite matrix	UC in graphite	UO_2 in BeO matrix	UO_2 in BeO matrix	UO_2 in graphite matrix, tubular shape, 5 1/2 in. long, 1 in. OD, 1/2 in. ID	UO_2-BeO	UO_2
	2.2 (feed enrichment)	1.75	~11[a]	~13[a]	Natural	Natural	1.19	6.8[a]	5.2[a]	48.4[a]	20	8.96	93.1	93	~62	30
sphere	Type 304 stainless steel	Stainless steel	Graphite can	Graphite can	Magnox	Al2 Magnox	Beryllium	Graphite	Graphite or Si-SiC	Graphite	Graphite	Hastelloy-X or type 316L stainless steel	Hastelloy-X		Hastelloy X	0.006-in.-thick type 318 stainless steel
	0.750 in. OD, 0.020 in. wall	0.415 in. OD, 0.015 in. wall	4.15 in. OD, 3.14 in. ID (graphite can)	0.375 in. OD, (graphite can)	0.630 in. OD, 0.072 in. wall, finned		0.653 in. OD, 0.0730 in. wall, finned	None	None	None	None	0.375 in. OD	0.241 in. OD, 0.030 in. wall	None	0.375 in. OD, 0.020 in. wall	12 concentric cylinders, 0.045 in. thick, 1.85 in. OD, 36 in. long
	None	None	None	None	None	None	None	1.5	2.5	1.5	2.36	None	None	None	None	None
	16	15	9	3.52	31	49	122	9.0	20.7	1.5	9.8	6.37	1.83	6.3	~2	2
	19	14	7.5	5.25	21	25	12	8.1	12.4	8.1	11.8	6.37	1.83	3.3	6.3	2.3
,	61.2		242	396	11	20	113	672	187	710	56.7	373	700	31.1	6.7	300
ft^2	66,800	~45,000	87,400	76,100	60,500		76,600	102,600	35,400	81,500	31,700	77,200	78,100		77,100	33,160
	4310		4490	76,100	10,300			11,200	73,200	417	5270	3565	127			
	510	482 to 617	662	662	284	324	550	550	550	550	392	745	791	1600	750	800
	300	270	294	294	100	185	541					1106	298		735	242
	1050	932 to 1067	1382	1382	636	707	1050	1250	1350	1250	1560	1300	1200	2400	1300	1200
	292			279	94.5		511	965	700	1000	132	1065	274	485		204
	4.27×10^6	2.74×10^6	4.8×10^5	76.3×10^3	424×10^6	216×10^6	3.62×10^6	1.34×10^6	2.75×10^6	4.32×10^4	3.71×10^5	3.74×10^5	8.96×10^4	10,250	60,000	63,000
ra-	1080		>2200, in sleeve 2700, in can 3350, in spine		750	840	~110 ~230				39.4	>1000	156 190		240	190 300
surface	1575 (max.)	1112 (max.)	2000 (max.)	2462 (max.)	766.4 (av.)	820 (av.)	1350 (max.)					1500 (max.)	1750 (max.)		1500	1650
										1200						
								2170	2000	2000	1650					
il	~3000		3200				~3000	2440	2200	2200	2550		2160 (UO_2-BeO pins) 2650 (UO_2 pins)			

Table I.1 (continued)

EGCR	AGR	HTGR	Dragon Project	Calder Hall, One Reactor	Hinkley Point, One Reactor	FWCNP	Sanderson & Porter Pebble-Bed Reactor	ORNL Pebble-Bed Reactor (PBR)	ORNL Pebble-Bed Reactor Experiment (PBRE)	Brown Boveri and Krupp (AVR)	MGCR	ML-1	UHTREX (Turret)	EBOR	GCRE-1
900	850	1005	Heat sink	590	655.2		1000	1050	Heat sink	941	Closed-cycle gas turbine	Closed-cycle gas turbine	Heat sink	Heat sink	Heat sink
1265	650	1450		200	640.0		1450	2415		1029					
20.66 ft 4 in. 46.33 ft	21 ft ID 2.88 in. 53.5 ft	14.33 ft 2 in. 34.5 ft	11.5 ft 2.25 ft 58 ft	41 ft 2 in. 70 ft	74-ft (sphere) 3 in.	Pressure tubes	13.75 ft 0.458 ft 21 ft	32.1 ft (sphere) 4 in.	9.25 ft 0.271 in. 25 ft		10.38 ft 17.9 ft	Pressure tubes	14 ft (sphere) 1.75 in.	116 in. 280 in.	Aluminum pressure tubes, 1.875 in. OI
Carbon steel	Aluminum-killed mild steel	SA212-B steel		Steel	Steel	None	SA212-B steel	SA212-B steel	SA212-B steel		SA302-B steel	Type 321 stainless steel	SA212-B steel	A387-B steel	
21 Stainless steel clad B_4C	25 Boron steel and stainless steel	36[c] B_4C+C in stainless steel	24 B_4C	104 Boron-stainless steel		33	12 Haynes-25	20 B_4C in stainless steel	Inconel		16 to 32 Rare earth in stainless steel	12 (blades) Cd-In-Ag in type 202 stainless steel	5 ZrB_2 in molybdenum container	4 Dy_2O_3-Al_2O_3 in Hastelloy X	10 W-Cd-In
Winch type	Manual and manual/automatic	Combination hydraulic, mechanical, and pneumatic	Motor and winch	Motor and winch		Rack and pinion	Rack and pinion	Motor and winch	Motor and winch						Motor-driven nut and bolt
1-1 Stainless steel		3-1 Carbon steel		1-6 Steel			2		2 Steel		4 Steel or iron				
- 11,200, UO_2	13,200, UO_2	190, U^{235}; 1190, Th^{232}	20, U^{235}	132, natural U	415, natural U	152, U^{235}	90.2, U^{233}; 992.2, Th^{232}	625, U^{235}	21, U^{235}	23.5, U^{235}; 340, Th^{232}	160, U^{235}; 2250, U^{238}	49, U^{235}	6.8, U^{235}	100, U^{235}; 60, U^{238}	19.2, U^{235}
~10,000	~10,000	~80,000		~3000	~3100	~10,000	48,500				~30,000	~8,000		2000	
None	None	2760:8.54:1	2128:6.83:1	None	None	None	3745:11:1	4000:18:1		None	None	None	None	None	None
	1.5×10^{13} (max.)	2.75×10^{13}	4.5×10^{13} (max., initial)				1.3×10^{4} (initial)					1.9×10^{12}	2×10^{13} to 4×10^{13}		
						4×10^{-4}		0.3×10^{-3}	1.5×10^{-3}		2×10^{-4}	3×10^{-5}			
14		~40		5.3	4.1	9	16.9	8.38	5.86			4.3			
18	14	25[d]	17			19.4	17.3	11.1	14		49	6.9[e]	~29	25	20.7
al 2.5 $\times 10^{-4}$		$<1 \times 10^{-3}$	0.6 in./sec	2×10^{-6}		1×10^{-4}					2×10^{-4}				
2	4	2	6	4		2	3	2	1		1	1		1	1

^{35} + Th^{232} initially present.

nd height.

ctrically-driven emergency rods.

down ~18% reactivity.

nt for another 2.2%.

Group (FWCNG)[16] reactor had a parallel interest in this type of fuel cladding. The Army GCRE,[17] recently operated and dismantled, used a concentric tube fuel system containing a dispersion of UO_2 in stainless steel, and the more recent ML-1[18] and the beryllium oxide reactor experiment[19] (EBOR) use a mixture of UO_2 and BeO in Hastelloy-X tubes. The central-station power reactors use low-enrichment fuels, that is, enrichment in the range of 1 to 3% U^{235} and have an upper cladding temperature, based on the creep-rupture strength and corrosion resistance of the material, in the neighborhood of 1800°F.[20]

3. Purged thorium-uranium-graphite fuel systems are to be used in the U.S. high-temperature gas-cooled reactor (HTGR)[21] and the European Dragon reactor.[22] The fuel consists of a graphite matrix containing thorium and uranium carbides that is made up in cylindrical shapes about 3.5 in. in diameter with an impermeable outer shell of graphite and a provision for purging the fission-product gases released from an inner controlled-permeability fuel body. The upper surface temperature limit of the fuel is fixed at the level where the impermeability of the graphite[21] is lost and is in the range 2000 to 2400°F.

4. Unpurged uranium and thorium-uranium-graphite fuel systems are to be used in the Brown Boveri and Krupp AVR reactor[23] at Jülich, Germany. The AVR is the principal power-reactor system being actively developed that uses this type of fuel, although studies of comparable pebble-bed reactors were carried out by Sanderson and Porter[24] and ORNL.[25-27] These fuel systems differ from the purged-fuel system mainly in that the fuels to be used are expected to contain the bulk of the gaseous fission-product activity in a low-permeability graphite shell,

with possibly a pyrolytic-carbon coating around the fuel particles. The United States UHTREX[28] system at Los Alamos is also included in this category, although only a limited effort will be made to hold the fission-product activity within the fuel body of UHTREX. The fuel considered for pebble-bed designs is in the form of spheres ranging from 3/4 to 1.5 in. in diameter. The tubular-shaped UHTREX fuel is 5 1/2 in. long, 1 in. OD, and 1/2 in. ID. The upper temperature limit of these fuel bodies is similar to that for the purged fuel.

Moderators. The moderator materials presently available for gas-cooled reactors consist of graphite, beryllium oxide, heavy water, and light water.

1. The Magnox systems, the AGR, and the EGCR all use a graphite-moderated and -reflected core structure with the fuel inserted in cylindrical graphite channels through which the cooling gas flows. The Dragon and HTGR incorporate the moderator into a homogeneous fuel body but provide a graphite reflector, the flow channels being established by the natural spacing between the cylindrical bodies. The AVR has a similar, homogeneous, fueled core with a graphite reflector, but the flow channels are through voids in the randomly packed fuel spheres. In the UHTREX, the fuel bodies are inserted in discrete channels of graphite.

2. The EBOR uses a beryllium oxide moderator in a geometrical arrangement somewhat like that of the EGCR.

3. The FWCNG reactor, now discontinued, was to use a D_2O-moderated system, with cooling gas flowing through pressure tubes containing the fuel in a callandria surrounded by the heavy water.

4. The GCRE and ML-1 are light-water-moderated systems with pressure tubes containing the fuel. The design is comparable to that of the FWCNG system.

Coolants. Several coolants for gas-cooled reactors have been considered.

1. Carbon dioxide is used as the gas coolant in Magnox, AGR, and FWCNG designs at pressures in the range of 200 to 300 psig. The upper temperature limit of the gas, as determined by compatibility with the moderator and fuel bodies, has been fixed at around 770°F for the Magnox fuels and 1050°F for the AGR fuel.

2. The EBOR, AVR, Dragon, and UHTREX all use helium as the coolant to extend the high-temperature limits by taking advantage of the coolant's compatibility with all materials of construction. Impurity limits have a nominal effect on the temperature capability, but, even so, the EGCR was designed for 1200°F, the HTGR for 1380°F, the EBOR for 1500°F, the AVR for 1560°F, and the UHTREX for bulk coolant temperatures of over 2000°F.

3. Nitrogen is uniquely used as a coolant for the military GCRE and ML-1 reactor systems. The 1200°F bulk gas operating temperature appears to be somewhat arbitrary and not obviously associated with any material problems.

4. Hydrogen and steam have also been considered for gas-cooled reactor coolants, but none of the reactors currently under consideration use them in the design. For this reason no working limits have been established for them.

Power Recovery. Both indirect and direct power-recover cylces have been considered.

1. Virtually all central-station power plants currently under design use steam-condensing cycles for power conversion, with steam conditions provided in the range of 700 to 950°F and pressures up to 1400 psi. No designs of very high-performance steam systems of 2400 psi and 1000°F are being executed.

2. Designs for direct power conversion by gas turbines are presently limited to the military reactors and studies of maritime application. Only the ML-1 designers have plans to couple a gas turbine directly to the gas coolant, although a study of this application was made for the predecessor of the EBOR, the MGCRE, which was a maritime design. In the ML-1 the gas temperature limit is set at 1200°F for the gas turbine.

Safety Characteristics

With the wide variety of design features mentioned above, no highly specific safety characteristics can be shown to cover all gas-cooled power-reactor systems. The United Nuclear Corporation[29-31] has made a study of the various reactor concepts and has categorized[31] the safety features into the following two groups: (1) those that adversely affect safety and (2) those that enhance safety.

Characteristics that Adversely Affect Safety. Relative to other types of civilian power reactor, gas-cooled power reactors have several disadvantages insofar as reactor safety is concerned. There is no inherent mechanism for the safe removal of afterheat following depressurization or loss-of-flow accidents. In most instances, free-convection

cooling is insufficient. Unlike liquid coolants, the gaseous coolant cannot remove heat through a phase change.*

The fuel cladding is more susceptible to failure during temperature transients. In gas-cooled reactors the cladding normally operates at higher temperatures than in other reactors. The higher temperatures are the result of the relatively low heat transfer coefficients of gaseous coolants and the desirability of achieving good cycle efficiencies. During a power transient, therefore, the cladding temperatures in gas-cooled reactors are considerably higher than in other reactors, and therefore the probability of cladding failures is greater.

In the event of a large reactivity insertion, the absorption of energy by the coolant does not aid in shutting down the reactor because of a weak coupling between the coolant density and the reactivity.

Afterheat removal in the event of a loss of coolant accident is complicated by the possibility of chemical reactions of core materials with inleaking air. This is particularly true in the higher power-density designs where the core materials operate at relatively high temperatures.

Characteristics that Enhance Safety. A number of inherent features that increase safety are common to many gas-cooled power reactors. The prompt temperature coefficients of reactivity are negative because fuels containing large quantities of fertile materials have strong negative Doppler effects. These prompt negative temperature coefficients tend to limit the transients caused by accidental reactivity insertions.

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*A severe depressurization accident in a light-water reactor can remove a sufficient amount of water from the system so that only steam is immediately available to cool large portions of the core. In this instance, heat removal by a coolant phase change is of little consequence.

The core power densities are relatively low compared with those of other power reactors. This reduces the problem of afterheat removal following severe loss of coolant accidents.

The energy stored in the gaseous coolant is low compared with that stored in high-pressure liquid coolants. The small amount of stored energy relieves the requirements on the secondary containment vessel, and thereby reduces its cost.

Large core heat capacities are available because of the presence of substantial quantities of moderator materials. Large heat capacities, in effect, help reduce the temperature peaks during power transients.

The neutron lifetimes are relatively long, because of the types of moderator selected, the low enrichment fuels employed, and the moderator-to-fuel ratios used. This means that for a given reactivity insertion above prompt critical, the periods are relatively long, and hence the resulting excursion is more easily terminated.

There is generally good compatibility of the coolant with the core and primary loop materials at the operating conditions of the system if impurities, such as water vapor and air in the coolant, are kept at low levels. There is no significant Wigner-stored energy for graphite-moderated gas-cooled reactors operating with graphite temperatures in excess of 600°C. This includes all graphite reactors examined, except the Calder Hall reactors.

Removal of moderator by evaporation, sublimation, or burning reduces reactivity. Further, reactivity is insensitive to changes in coolant density, such as in a cold coolant accident.[23]

The characteristics discussed above that are referenced primarily to water-cooled reactors deserve some clarification as they affect the vaious designs discussed here. Unlike the water reactor systems, the gas-cooled reactors do not have available the latent heat of vaporization of water as an afterheat sink, but they have, instead, a large mass of solid moderator with a substantial heat capacity. If the moderator can be allowed to go to rather high temperatures without exceeding the fuel temperature limits, its capability for absorbing heat from the fuel can be enormous (approaching or exceeding the water system heat-absorbing capability). In reactors of the Magnox type, where the melting point of the cladding is relatively low (580°C, 1075°F), there is very little available heat capacity. The situation improves substantially for the UO_2-fueled systems, where hot-spot temperatures are permitted to reach 1800°F. The condition is less clear in the case of graphite-matrix fuel bodies, but presumably they have substantial emergency heat capacity because the average temperature of the fuel body is well below the upper temperature limit for catastrophic failure. Regardless of whether it is gas-cooled or water-cooled, a large power reactor will overload this afterheat sink unless some supplemental heat-removal means is provided. Any important differences between systems will be associated primarily with the amount of time available in which to supply the supplemental heat-removal capability.

Since the cladding of fuel elements in gas-cooled reactors operates at high temperatures compared with cladding in water systems, there is an implied low safety margin against fuel failure during transients. The factors leading to failure are so different in the several systems

that it is easy to be deceived in this matter. Water-cooled systems, for example, with their very high power densities may undergo severe fuel temperature transients if a flow channel is partially blocked and the fuel body becomes blanketed with steam. A similar case may be experienced with the Magnox fuels, which operate very close to the cladding melting point, and to a lesser degree in the stainless steel-clad and graphite fuels, which have a somewhat greater temperature margin. The safety margin is determined by the difference between the rate of heat generation and the heat-removal capacity.

Water-cooled and -moderated reactor systems have a unique safety characteristic in that the water expands rapidly with temperature and may, in fact, be removed by vaporization during a power excursion and thus stop the incident. Gas-cooled reactors, on the other hand, in common with most other solid-moderator systems, gain no advantage in this area. The graphite-moderated gas-cooled systems have only a slightly negative moderator temperature coefficient at the beginning of the reactor fuel cycle, and at equilibrium fuel cycle conditions the plutonium or U^{233} buildup may turn the moderator coefficient slightly positive. Because of the large heat capacity in the gas-cooled moderators, they have a very slow response to temperature transients, and the very large negative temperature coefficient of the fuel usually controls reactivity. The principal effect of the moderator coefficient is therefore confined to long transient periods in which the temperature of the moderator can be raised by energy release from the fuel or other undefined energy sources.

Unlike water coolants, gases have a negligible effect on the core reactivity. Consequently, removal of the coolant does not represent a means of limiting a nuclear excursion. Thus the only means of stopping an excursion is through the physical rearrangement of the fuel or moderator or insertion of a supplemental poison.

Somewhat reminiscent of the early concern for metal-water reactions in nuclear systems is the problem of dealing with secondary chemical reactions following an accident. The problems differ in type and degree in various gas-cooled systems and are listed in Table I-2. The rate and temperature characteristics of the reactions are such as to make them successively more rapid with higher operating temperatures, thus requiring progressively more stringent safety precautions.

Toxic Gaseous Releases

Aside from those fission-product sources released by the fuel, the purged-fuel systems establish through their fission-product trapping arrangement a second bulk source of fission products that requires safety consideration. High reliability in the trapping systems is an inherent part of the safety requirement for this type of reactor.

Inhalation hazards must also be considered in the case of the D_2O-cooled and helium-cooled systems because of the formation of tritium by radiation. The conventional problems of argon activation in the shield-cooling system must likewise be considered. In addition there may be other safety considerations from the activation of nitrogen in the ML-1 coolant or beryllium reaction products from the EBOR design. The limited information concerning these factors is not such as to warrant major concern for their hazards implication.

Table I-2. CHEMICAL REACTION HAZARDS IN GAS-COOLED REACTORS

Reactor Type	Chemical Reaction	Conditions for Event	Consequence
Magnox	CO_2-uranium	Cladding melts or breaks	Activity release from uranium oxidation
	CO_2-Magnox	Blocked channel causes hot spot of ~630°C	Runaway fire involving Magnox and uranium
	Air-graphite	Hot channel coincident with air inleakage	Runaway fire and gross activity release unless supplemental cooling is supplied
	Magnox-uranium-air	Hot channel and air inleakage	Runaway fire and gross activity release; graphite-air reaction probability increases hazard
UO_2 fueled	Air-graphite	Loss of cooling coincident with air inleakage	Runaway oxidation, cladding melting, UO_2 oxidation, gross activity release, and containment-can overpressure unless supplemental cooling is provided
	Steam-graphite	Loss of cooling coincident with steam inleakage; reaction rate slow at credible transient temperatures around 1200°F	Hydrogen generation could lead to explosion and containment breach unless reaction is stopped or hydrogen concentration is diluted below explosive limits

Table I-2 (continued)

Reactor Type	Chemical Reaction	Conditions for Event	Consequence
	Air-UO_2	Cladding defective; air inleakage; high UO_2 temperature	Gross activity release to containment vessel
Fueled graphite (all types)	Air-graphite	Air inleakage; reactor temperature at normal conditions	Runaway fire; gross activity release
	Steam-graphite	Moisture inleakage at normal reactor temperatures	Hydrogen generation could lead to explosion; low moisture inleakage can cause increase in fuel cladding fission-product permeability

British Public Safety Evaluation for Magnox Reactors

Normally the evaluation of reactor hazards is concentrated on assuring that the public safety cannot be unreasonably endangered by the release of radioactivity and fission products. The various designs require different approaches to the assurance of public safety. Some insight is provided to the British approach by the UKAEA Guide to Safety Report Requirements.[32] This guide is primarily related to the Magnox reactors for which no secondary containment is required other than that normally provided by the biological shielding around the reactor. For this reason the British pay a great deal of attention to the analysis of accidents that might simultaneously release activity from the fuel and rupture the primary coolant circuitry. The British standards require the designer to show the following:

1. The control rods must be able to enter the core at any time when shutdown of the reactor is required. To show this, complete information on the structural design and heat-removal capability of the reactor core as it affects control rod insertion is required.

2. Since the Magnox reactors operate at temperatures very close to the melting point of the Magnox cladding and reactions can occur between carbon-dioxide and uranium when the cladding is not interposed, complete information on the thermal performance of the fuel channel and substantiation of the fact that partial or complete blockage of the fuel channel coolant flow cannot occur are required.

3. The more advanced Magnox reactor systems utilize on-stream fuel handling and the problems of fuel failure during fuel handling require

that these critical operations be examined for circumstances like those of item 2 above.

4. The primary coolant circuitry must be shown to have a very high level of reliability, and complete structural evaluation of the pressure-containing members, together with safety relief protection, must be presented.

5. Because the structural integrity of the system, regardless of the design, is predicated on assurance that fault transients will not exceed certain design limits, a complete summary of the control logic and control mechanisms must be provided to establish that protection can be guaranteed. Particular emphasis is placed on the kinetics of reactor shutdown to establish the time required for reactor trips under fault conditions.

The foregoing are required to show the adequacy of the design, but the designer is then required to evaluate typical faults that are arbitrarily established. Those listed in the AEA's Safety Report Requirements are considered to be typical and not necessarily all inclusive. They are:

1. The failure of the main blowers, considering first the case when the blowers do not provide any forced circulation, and then second the case in which pony motors permit the blowers to continue to operate at a reduced capacity but requiring, in the latter case, a time delay consistent with the time required to get the pony motors into operation. This accident assumes that the control rods scram and the Magnox can temperature does not exceed the 580°C melting point. The amount of time available for the shutdown action to occur is an important part of the evaluation.

2. Reactivity insertions in four instances: (a) continuous withdrawal of all the automatic rods at their fastest speed, (b) continuous withdrawal of the automatic rods at their most critical point with the primary blowers running at startup speed, (c) withdrawal of the automatic rods at full speed while the reactor is at full power, (d) complete withdrawal of one automatic rod and, as a corollary, a step increase in reactivity without compensation by the remaining control rod bank. The purposes of these analyses are to determine the limits of a nuclear excursion and to establish the time available for shutdown action.

3. The possibility of a fire occurring in a channel while the primary circuit remains intact is examined. Such an incident could arise from inleakage of air or a local hot spot that could permit reaction between carbon-dioxide and the Magnox or the uranium. The effect to be examined here is whether the fire could spread from a single channel through the entire core by the exothermal reaction between the Magnox and the coolant. This analysis is intended to show that such a fire can be extinguished by full flow circulation of the primary blower and to establish the length of time which such flow must continue to assure that the fire will remain extinguished.

4. The possibility of a fractured coolant duct in which the opening is sufficient to establish a coolant discharge coefficient of 1 is examined. In this accident the reactor is expected to be shutdown by the control rods, and examination is required of the effects resulting from complete loss of forced circulation and from the less stringent case when flow is reinstituted by the pony motors after some delay and

under conditions when the blower in the damaged circuit is inoperable. This accident analysis treats the case in which the reactor system is first filled with air and then, when the operator time permits, two valves in the fractured circuit can be closed and the air purged from the circuit with carbon-dioxide. Apparently the purpose of this analysis is to show that a runaway fire will not occur because of the reaction between the air and the fuel elements or the graphite before satisfactory operator action can be taken to control the accident.

5. Possible failures of the primary reactor vessel are examined for ultimate safety effects, but the design is expected to show that instantaneous fracture of the pressure vessel cannot occur. The purpose of this analysis is apparently to establish the ultimate limit of such an accident rather than to show that it is controllable.

6. The possibility of a fractured steam generator is considered and the effect of moisture on the reactor components is evaluated. In particular, the ability of the pressure-relief devices to hold the primary-circuit pressure limits within bounds must be shown.

These six accident conditions, under a variety of highly extenuating circumstances, are apparently utilized for the purpose of showing that gross quantities of radioactivity will not escape the primary coolant circuit. This is somewhat at variance with the U.S. approach, which starts with the assumption that some such accident has occurred, but the variation is mainly in the amount of activity that is assumed to be released during the accident. The U.S. designs permit more latitude in the activity releases from the primary coolant circuit because secondary containment provisions or very large exclusion areas supplement the public-safety protection.

U.S. Public Safety Evaluation

The U.S. methods of accident analysis can be illustrated by the hazards investigations of the HTGR and the EGCR. Although not standardized, the U.S. methods do, in large measure, parallel the British techniques for Magnox reactor analysis, but somewhat more severe accident assumptions are accepted for evaluation. The higher performance characteristics of U.S. designs as compared with the British Magnox systems probably justify this difference. It is observed that the United Kingdom's higher performance systems, the AGR and the Dragon, do provide similar safety features to those of the U.S. EGCR and HTGR designs.

The EGCR and HTGR designs consider accidents that involve the release of bulk quantities of fission products external to the primary coolant system. In the EGCR the activity release is associated with failure of a significant portion of the fuel cladding and release of gaseous fission products; however, the solid-fission-product retention capability of the UO_2 is shown to be essentially intact. In the HTGR the accidents considered assume the release of bulk fission activity not already purged from the fuel body to the trapping system or from the trapping system itself. In both reactors the containment of the activity releases must be provided by some means other than holding the primary coolant system intact.

These two U.S. gas-cooled central-station power reactors, the EGCR and the HTGR, provide steel enclosures for all of the primary coolant circuitry, including the steam generators. The design limits for the containment shells are fixed by the premises assumed in the accident analysis. The following are the important considerations: (1) leak

tightness with respect to activity released at the time of the accident or subsequent to it; (2) design pressure based on the pressure rise resulting from the release of primary system coolant, secondary gas reactions (such as air with graphite or steam with graphite), steam release, and the gas temperature rise resulting from the stored energy in the gaseous coolant and from the exothermic chemical reactions; (3) design temperature based on the temperature rise resulting from coolant release, afterheat, steam release, and exothermic chemical reactions; and (4) missile resistance to flying objects propelled by coolant stored energy, failure of rotating machinery, or hydrogen explosions.

The HTGR designers have elected to use a low-oxygen atmosphere within the containment shell in order to avoid the problems arising from hydrogen explosions and air-graphite reactions. The lower operating temperatures in the EGCR have permitted designers to cope with these problems by providing a nitrogen purge system that will be brought into action after an accident of major proportions.

An illustration of the type of analysis performed in the U.S. is present in the ORNL conceptual design study of the EGCR's emergency-cooling system.[33] The core thermodynamic behavior was examined to determine whether the following conditions could be met: (1) no runaway graphite oxidation; (2) no cladding temperature above 1800°F in order to prevent further cladding ruptures and to limit oxidation of UO_2 in ruptured elements; and (3) an exit coolant temperature of less than 1350°F. The accident considered was one resulting from the rupture of the primary coolant circuit and sequential failure of one steam generator, which would cause failure of the primary coolant blowers and

instantaneous failure of some fuel cladding. An interesting and important difference in the design features of the CO_2-cooled AGR and Magnox reactors compared with the helium-cooled EGCR system is that release of steam into the primary coolant circuitry in a large quantity could overload the EGCR blowers and cause damage and failure, whereas in the CO_2-cooled systems the introduction of steam reduces the load on the blower because of the steam's lower density than that of the pressurized CO_2. The studies by ORNL were intended to show that by controlling the oxygen content in the ruptured primary circuit through a nitrogen purge and by early operation of a small independent emergency blower system, the core temperatures could be controlled within limits that would prevent further activity release. Evaluations were made of the maximum and average power channel temperatures in the reactor to show that if the emergency coolant flow could be established within several hours after the accident the reactor core temperatures could be controlled sufficiently. The study also showed that the emergency-cooling loop was capable of reducing the temperatures to a very low level within 24 hr after the emergency cooling flow was established.

As a result of this study, supplementary cooling provisions were made in the EGCR for emergency use. A water-spray system was added to cool the external surface of the containment vessel, and an emergency cooling loop with a nitrogen purge to control the oxygen content of the reactor core after a system rupture was provided. These features of the EGCR eliminate the need to operate the steam generators following a loss-of-pressure reactor accident, in contrast to the Magnox systems which are shown to be safe on the premise that the bulk of the gas-circulation and

steam-generator equipment remains intact. This seeming conservatism in the EGCR design develops because, unlike the Magnox system CO_2 blowers, the EGCR blowers cannot be expected to survive a gross inleakage of steam, one of the accepted accident postulates.

The HTGR designers postulated[21] a similar accident for safety-evaluation purposes, but the safety solution is different and brings out some important differences between the clad-UO_2 and fueled-graphite reactor systems. In the HTGR the graphite fuel can tolerate very high temperatures, above 3000°F, without melting, whereas the EGCR fuel cladding would melt at such temperatures. Thus, during a loss of cooling accident, the reactor core of the HTGR can safely rise to very high temperatures. The arrangement permits the HTGR to release its afterheat through radiation to a water jacket surrounding the reactor vessel, eliminating the need to provide any forced gas circulation. This simplification is not without its penalties. The air-graphite and steam-graphite reactions must be avoided or restricted, and the HTGR design does this by specifying a low oxygen atmosphere inside the containment shell for routine operation to assure no oxidation of the graphite and no reactant for the hydrogen generated through steam release. The very high temperatures do pose a difficulty in keeping the core support structure intact, and this is solved by providing copper heat conductors to carry the heat from the core supports to the vessel wall, where it can be cooled by the vessel cooling jacket. The high temperature also gives rise to concern for the vaporization of fission products from the fuel, thus sacrificing shutdown margin from this source. Supplemental fused control rods are provided to offset this potential difficulty.

Reactivity accidents are evaluated on much the same basis in the EGCR and HTGR. Both designs are based on assuring a negative fuel temperature coefficient to guarantee a ceiling on any nuclear incident, but both rely on control rods to hold the reactor system permanently subcritical. Because the bottom-entry control rod system of the HTGR raises the classical questions about rod-drive disablement, the rod drives have been supplemented with emergency electrically driven safety rods inserted from the top of the reactor. These latter rods are designed so that they can be driven through the core even if the core becomes disarranged. The EGCR design uses gravity safety-rod insertion and only a fraction of its total of 21 rods is required to shut down the reactor.

The types of accidents considered in the safety evaluation of U.S. gas-cooled reactor systems normally arise from failure of major system components. A number of major components are examined as part of the safety analysis in a manner not unlike that specified by the U.K. Table I-3 lists the types of failure considered, along with the potential difficulties and the protective measures provided.

The protective features listed in Table I-3 are similar to those provided in U.S. water-cooled reactor systems. The most troublesome corrective measure is the location and isolation of a defective steam generator. Very prompt valve closure is required in order to prevent or minimize damage, and the inleakage of moisture is used as the isolation signal.

German Containment Practice

The Jülich reactor[34] offers an unusual containment arrangement that has some highly attractive safety features. The ability of a reactor

Table I-3. TYPES OF COMPONENT FAILURES ASSUMED IN SAFETY ANALYSES

Component Failure	Problem Areas	Corrective Measure
Primary power failure	Loss of core cooling	Natural circulation to steam generators under pressurized conditions (EGCR), or circulation by auxiliary-power-driven pony motors (HTGR) or emergency diesel power
Primary blower failure	Partial loss of core cooling	Reactor shutdown
Steam generator tube rupture	Over pressure, blower damage, and steam reactions with graphite	Relief valves, automatic steam generator isolation, reactor shutdown
Primary coolant system rupture	Loss of core cooling, fuel element failures	Reactor shutdown, blower circulation continued

system to cool itself by natural-convection flow to the steam generators is a very desirable asset, and many reactors are designed to provide natural-convection cooling during a normal shutdown with the primary coolant circuit pressurized. Most gas-cooled systems lose this capability when depressurized, but the reactor at Jülich has been ingeniously designed to retain this capability. The Jülich design provides two containers, with the inner one capable of holding the normal design pressure; the outer larger vessel is designed for a lesser pressure. A rupture of the inner container releases the reactor contents to the outer one, which is filled with an inert gas. The pressure rise of the outer container is such as to assure sufficient gas density to continue the natural-convection cooling capability of the reactor system so that no forced circulation is required to protect the reactor internal components, even under the worst accident condition. While this arrangement does not establish a level of safety for this reactor that exceeds that of other systems under design, it does provide a method of afterheat removal following a major accident that eliminates the need for operating gas circulation equipment within the containment.

Safety System Provisions

All reactor safety systems are used to protect the plant from malfunctions resulting from the component failures and control system difficulties. A comparison of several gas-cooled reactor safety systems is presented in Table I-4. All the reactor systems considered provide for shutdown of the reactor upon loss of heat-removal capability. A second scram is provided for coolant overtemperature. In some sense

Table I-4. SIGNALS FOR EMERGENCY SHUTDOWN IN GAS-COOLED REACTORS

Fault	Signal					
	EGCR	FWCP	HTGR	Dragon	MGCR	ML-1
Loss of heat-removal capability	Flux minus flow	Low coolant flow	Not specified	Anomalous coolant temperature, pressure, and flow	Loss of coolant or sea-water flow	Not specified
Coolant over-temperature	High coolant outlet temperature	High coolant outlet temperature	High coolant outlet temperature	Anomalous coolant temperature	High outlet gas temperature	High outlet gas temperature
Loss of coolant pressure	Low coolant pressure	High rate of pressure change	Low coolant pressure	Anomalous coolant temperature	Not identified	Rate of pressure loss
Steam generator rupture	High coolant pressure	High rate of pressure change	High coolant pressure	Anomalous coolant temperature	Not applicable	Not applicable
Startup accidents	High flux in startup range	High flux level	Short reactor period	Excessive rod withdrawal rate	Not identified	Short reactor period
Turbine failure	None	None	Excessive steam pressure turbine trip	Anomalous secondary flow or pressure	Turbine overspeed	Not identified

Table I-4 (continued)

Fault	Signal					
	EGCR	FWCP	HTGR	Dragon	MGCR	ML-1
Loss of electrical power	Loss of control rod drive clutch power	Not identified	Not identified	Loss of electrical supply	None	Loss of electrical supply
Loss of instrument power	None	None	Loss of failure-free power supply	None	None	None
High activity release	None	Excessive radiation level	Excessive activity in containment volume	Excessive activity in stack	None	None
Vessel nozzle over-temperature	Loss of nozzle annulus flow	None	None	None	None	None
Control rod drive failure	Loss of nozzle interior flow	None	Loss of control rod hydraulic fluid pressure	None	None	None

Table I-4 (continued)

Fault	Signal					
	EGCR	FWCP	HTGR	Dragon	MGCR	ML-1
Loss of secondary coolant	Low steam drum level	None	None	None	Loss of heat exchanger water, loss of sea water	Loss of moderator flow
Fission-product trapping system failure	None	None	None	Loss of liquid nitrogen	None	None
Moisture inleakage	None	None	High moisture level	Not identified	None	None

this is a duplication of the loss of heat-removal capability, but the two incidents are not necessarily coincident, and an excessive coolant temperature could lead to rupture of the primary coolant system and a depressurization accident.

Since depressurization of the high-pressure gas coolant would cause a very substantial loss in the heat-removal capability, it could lead to very high core temperatures which would not be indicated by either of the two previously mentioned scram conditions. The signal used for shutting down the reactor in this case varies for the several systems examined here; low coolant pressure and rate of pressure change are both used as a signal. The indirect cycle systems also provide for a scram as a result of gross inleakage of moisture. This is normally associated with the rupture of a steam generator, and the signal provided for shutdown is high coolant pressure or high rate of pressure change caused by release of the higher pressure steam into the primary coolant system.

All the reactor systems provide protection against startup accidents by some type of high flux signal in the startup range or a short reactor period. There is little information about the kinetic behavior of the reactor systems under these conditions, but it appears that the protection is provided because low-power operations are normally performed with little or no heat-removal equipment in operation. Under these conditions the reactor system must rely on its inherent heat capacity for protection against power excursions. The startup accident protection thus prevents this limited heat-removal capability of the reactor system from being overridden by an accident during low-power operations.

It is noted that, with the exception of the EGCR, most reactor systems require emergency shutdown upon failure of the turbines. This omission is deliberate in the EGCR because a full-capacity bypass turbine condenser is specified in the design, and it is expected that full-power operation of the reactor will sometimes be carried out with the turbine out of service.

Loss of electrical power causes scram of the reactor in the case of the EGCR and the Dragon. The same result is probably obtained in the HTGR through an indirect signal, although this is not obviously the case. It is noted that the HTGR also provides emergency shutdown on loss of "failure-free" power. In many instances this is a superfluous signal, since loss of electrical power to the instruments will cause the reactor to behave as though one of the other scram symptoms has occurred.

Of the several designs, only the EGCR does not provide scram on high-activity release. This signal is an indication of trouble and requires some emergency action, but in the case of the EGCR it is apparent that the difficulty must be recognized at a much earlier stage in the accident than the activity release. On the other hand, the HTGR and the Dragon do provide shutdown because of the presence of excessive activity in the containment volume or the containment stack. This may be the result of anticipated difficulties in the fission-product trapping system that would not be apparent from the operations of the primary reactor system but would warrant immediate shutdown in preparation for assurance of public-safety protection.

The EGCR safety system provides for a scram of the reactor upon receiving a symptomatic trouble signal from any of three pieces of equipment.

These are loss of the nozzle annulus flow, indicating potential overtemperature of the reactor vessel nozzles; loss of nozzle interior flow, indicating that cooling for the control rod drives has been lost; and low steam-drum level, indicating that the steam generator tube sheet may be in danger of losing its water coverage and thus being subjected to high thermal stresses. These equipment protection features are provided to avoid more serious accidents. In the case of either the vessel nozzle overtemperature or the highly thermal-stressed tube sheet, a reactor loss-of-coolant-pressure accident could follow. In the loss of control rod drive cooling case, the principal interest is to assure that the rods are inserted before the heat from the reactor has time to overheat the drive mechanism. In none of these cases does the symptom indicate instantaneous reactor trouble, and the safety action is intended to avoid operating difficulties that may occur minutes or hours later.

The two reactor systems using vented fuel elements, the HTGR and the Dragon, appear to have somewhat different philosophies regarding problems resulting from failure of the fission-product trapping system. Both reactors, as noted previously, are scrammed on indication of activity release to the containment shell, but the Dragon provides an additional scram on loss of cooling in the fission-product trapping system. The inference drawn is that prompt shutdown of the reactor can in some way relieve the hazard associated with this failure for the Dragon. Such is not the case for the HTGR, but it is most probable that the action required for the Dragon is associated mainly with relieving the operator of a decision responsibility during this emergency.

From the presently available information, the HTGR appears to be the only reactor requiring scram on high moisture level. This action does not appear to be a problem of public-safety protection, but rather one of protecting against fuel damage by shutting down the reactor system. It is presumed that the reactor core could be cooled to a temperature level at which moisture would not attack the low permeability fission-product barrier and thus prevent fuel damage leading to outleakage of activity into the primary coolant circuitry. Public safety would be endangered only if this incident occurred concurrently with a loss of coolant pressure and a breach of the reactor's secondary containment system.

Summary

The progress in the safety evaluation of gas-cooled reactor systems in the United States is not yet such that accepted standards for safety are established. It does appear that British and U.S. designs are on comparable safety bases where equivalent reactor designs are involved, but more liberty is taken in the containment of the Magnox systems because their lower power densities simplify the heat-removal problems under accident conditions. The U.S. power reactor safety evaluation standards developed for water-reactor systems appear to be adaptable to gas-cooled systems, with the exception that nuclear shutdown can be permanently assured in the latter system only by supplemental poison insertion. Some compensating features of the gas-cooled system, including the lower quantity of stored energy in the coolant, the higher temperature capability

of some fuels, and the lower reactor power density, have advantageous effects on the over-all system safety. (M. Bender)

MEDICAL ASPECTS OF BERYLLIUM POISONING

Inasmuch as beryllium is being used in increasing quantities in the nuclear energy program, it continues to be of interest to nuclear safety. A previous article[35] in Nuclear Safety reviewed the early history of beryllium toxicity, discussed some of the symptoms of the various forms of the disease, and presented the recommended criteria for the control of beryllium hazards in industrial situations. The present article elaborates on the medical aspects of beryllium toxicity and what is being done about the prevention and treatment of beryllium disease.

The story of beryllium disease is "one of the most frustrating, contradictory, infuriating and controversial episodes in medical history."[36] Although some still disagree, the available evidence[37-39] supports the general agreement that beryllium toxicity is a reality and has been the cause of death of a number of people. Much of the difficulty in recognizing the disease lies in the fact that the various clinical patterns resulting from beryllium exposure are not unique in terms of their body effects and are no different than those produced by other intoxications or infections. There is, further, no clearcut distinction between the so-called acute and chronic disease. There appears to be, instead, a spectrum ranging from reactions as a result of massive exposures, which produce a response within days, to those that are so insidious that the reactions take years to develop and yet may be ultimately fatal. Tepper, Hardy, and Chamberlin[37] have arbitrarily defined acute beryllium disease as that including beryllium-induced disease patterns with less than one year's natural duration and have labeled cases

of more than one year's duration as chronic beryllium disease. They point out that there is often an overlap between the two, but, in most cases, acute forms of beryllium intoxication are of several months duration, whereas the chronic disease, for all practical purposes, is never entirely resolved.

Acute Beryllium Disease

The acute disease patterns were the earliest recognized manifestations of the toxic effects of beryllium on human tissues. With the advent of control measures, this type of disease has almost disappeared. The target organs of the acute disease are the skin and the mucous membranes of the eyes and respiratory tract. In all syndromes there is a dose-effect relationship, that is, the more intense or prolonged the exposure, the greater the toxic response. The skin manifestations and the inflammatory reactions of the conjunctivae and membranes of the upper respiratory tract are, relatively, not serious, and recovery has been the rule. Acute pneumonitis has been by far the most serious of the acute syndromes and takes two general forms. One is a fulminating illness following brief but massive exposures that usually occurs within several days of the exposure. The second is a more insidious disease that usually shows up after prolonged exposure to lesser concentrations and, at times, requires months to become manifest. The clinical features of these intoxications resemble those of pneumonias induced by other agents, and treatment is essentially the same. In the majority of documented cases, the course of the disease was relatively benign, and total recovery followed within one to six months. Fatalities have also been

documented, but these occurred in the early years before beryllium toxicity was fully appreciated or recognized. Although present-day controls and preventive measures have made the acute disease a rarity, accidental exposures still occur, and it is likely that cases of this nature will continue to be seen in the future.

Chronic Beryllium Disease

The chronic disease was recognized later, chronologically, than the acute disease, and it has been by far the more controversial of the two. There are a number of reasons for this. The clinical characteristics of chronic beryllium disease are no different from those of some other not well-understood diseases, such as sarcoidosis and miliary tuberculosis. There is apparently a delay in onset of the illness following exposure, and there is little correlation between the degree or intensity of exposure and the severity of disease.

The clinical character of the chronic illness differs from that of the acute in several aspects. First, it is a systemic disease; second, there is no cure; and third, it is commonly progressive in severity despite removal of the person from the exposure. It is sometimes surprising to those who are aware of the toxic effects of beryllium that the chronic disease is a systemic disease that involves several organs and systems rather than a disease that is localized in the lungs. It is true that beryllium exerts its most deadly effects on the lungs, but the same type of response as in the lungs has been found in many other body tissues, including the skin, liver, spleen, kidneys, skeletal muscles, lymphnodes, and bone marrow. Berylliosis differs in this regard

from pneumoconiosis, a disease with which it is commonly compared but which is essentially a dust disease localized in the lungs. It is quite possible that some of the systemic manifestations contribute to the disability of the person who has the disease. There is, unfortunately, little more known about this today than when it was first recognized.

Lack of correlation between intensity of exposure and severity of disease has always been a bone of contention in discussions of the chronic disease. Many cases do show a direct relationship, and yet it is not possible to discount the number of cases that have resulted in fatalities or severe disability despite histories of minimal exposure, some even after the cessation of exposure and years after such cessation. "Ignorance of exposed population numbers, fluctuations in plant populations, and uncertainty in the degree and nature of atmospheric contaminations make correlations very unsatisfactory."[37] This varying dose-effect relationship is not unique, however, in medical histories, and the same type of response has appeared in other diseases, for example, infectious hepatitis. The lack of agreement between dose and effect has led some observers to postulate an immuniologic or allergic type of reaction as the basic body response in the chronic disease. Experimental and clinical proof has not been obtained to substantiate this theory, and the basic mechanisms are no clearer than they were some 15 years ago.

What is the current status of treatment of chronic beryllium disease? The evidence accumulated in the past 15 years indicates that Cortisone and the more recent steroids favorably affect the course but do not cure the disease.[40] Prior to the advent of the steroids, treatment of the chronic disease was restricted to supportive measures and

was generally ineffective. Except for the patient who fortunately had a mild form of the disease, the outlook was poor. Patients with berylliosis almost invariably got worse with time and ultimately became respiratory cripples or died from pulmonary or myocardial insufficiency. Steroid therapy has changed the prognosis so that a person with chronic beryllium disease has a much better outlook in terms of milder symptoms, less disability, and fewer complications; however, there is, as yet, insufficient knowledge of the long-term effects of therapy to conclude that cures are obtained.

Chelating agents have shown promise in treatment of chronic beryllium effects in experimental work on mice and other animals.[36] Evidence has been lacking that they are of value in the treatment of human chronic beryllium disease.[37] In the few actual cases in which they have been used, other variables have clouded the circumstances, and objective evidence of improvement has been scanty or absent.

What is the mechanism of action of the chronic beryllium disease? This still is as little understood as it was in the early days when the disease was first recognized. It is known that the primary lesion is a granulomatous response (much as seen in tuberculosis and sarcoidosis), but clinical patterns have shown extreme variability. Most appear to show some progressive disability. There is a progressive loss in respiratory function, and death, when it occurs, is usually due to a complication affecting the cardiovascular system (cor pulmonale). Not infrequently the terminal event is an infection with which the weakened lung cannot cope. The same reactions are seen in silicosis. It is important to point out that the presence of beryllium as such in the

body does not establish the disease, and neither does the fact that a granulomatous reaction can be demonstrated, since the same response can occur in other diseases. Diagnosis is established when there is a clear documentation of beryllium exposure in addition to objective manifestations of the disease process in body tissues, that is, positive beryllium assays along with histological evidence of a granulomatous reaction.

Do acute cases of intoxication become chronic? Some apparently do; most probably do not. Of the known acute cases, one tenth have become chronic; but this is only an estimate because of the wide variation in clinical responses, lack of exposure data, etc.

The Patch Test

What is the current status of the patch test? This has remained a controversial subject since its introduction 15 years ago. Curtis,[41] who did most of the early work with the test, showed that small amounts of inorganic beryllium salts gave a positive skin response in patients with beryllium disease. He did not claim that this necessarily reflected other organ sensitivity. Furthermore, a negative test did not necessarily mean that beryllium disease was not present. In a later series of cases,[42] Curtis found the test positive in 32 patients with beryllium disease, and negative in 14 with other chest diseases. He points out, however, that he considers a positive test "not in itself pathognomic of a pathologic process other than that in the skin or mucous membrane," and that "the patch test could not stand alone as an absolute pathognomic sign for the presence of beryllium granulomatosis in the lungs." He further found that spontaneous flareups did occur in

control persons and considered that these indicated eczematous hypersensitivity induced by the test. Because of this latter finding, a number of physicians, notably Waksman,[43] feel that the patch test is potentially dangerous in a person whose tissues might contain berryllium if he does not yet have the disease or is in the state of clinical remission.

Hence its use in routine testing of beryllium workers and in screening persons who are to be exposed to beryllium is best avoided. The patch test is useful as a diagnostic tool, but its value is limited, since a positive reaction is evidence only that the subject is sensitized to beryllium and the reaction does not prove the existence of past or present disease. A negative test, however, in a person suspected of having beryllium disease makes the diagnosis less likely, since a negative reaction in the presence of disease is rare.

Toxicity of Different Beryllium Compounds

Have all beryllium compounds known been found to be toxic? Of all the beryllium compounds encountered in industry, only one, beryl, the natural ore (a beryllium aluminum silicate), has not been associated in any way with beryllium disease, acute or chronic, industrial or experimental. This is surprising when it is considered that much of the work involved in mining and handling the ore is dusty, and undoubtedly exposure has been sustained. All other commonly encountered beryllium compounds have been incriminated. The acid salts, beryllium sulfate and fluoride, primarily used in the extraction industry, have been associated with the acute disease in all its forms. Exposures to dusts and fumes of metallic beryllium, as well as to dusts of beryllium oxide,

have also been found to cause the acute disease. It may be that the reported exposures to metallic beryllium were actually exposures to the oxide, for fine particles of the metal certainly have a thin surface coating of oxidized material. Coarse particles or formed shapes of the metal itself are not toxic. The toxic properties of beryllium oxides have been shown to differ, depending on their method of manufacture.[44] The oxides prepared at relatively low calcining temperatures have produced cases of acute disease; the high-fired oxides have not.

It has been felt that physical factors are the reasons for the differences in behavior. It is known that the total surface area of the oxide particles increases with decreasing temperature and that the low-fired oxide is more rapidly soluble in water. How these factors relate to their disease-inducing properties has not been satisfactorily explained.

The greatest number of cases of chronic disease have been related to beryllium-containing phosphors, which were at one time used extensively in fluorescent-lamp manufacture. It was suggested by VanOrdstrand and others[45] that beryllium oxide was possibly the etiologic agent "in most, if not all, cases of chronic beryllium disease." This reasoning is based on the fact that the early phosphor workers and those exposed to finely divided metallic beryllium were also exposed to the oxide, and, in spite of the predominance of acid salts in extraction procedures, the oxides, as well, are present in most of these operations.

Incidence and Prevalence

What proportion of people exposed to beryllium contract disease? This, unfortunately, is still a question that cannot be adequately

answered, since data on incidence and prevalence rates are confusing. There is clear evidence that there is a threshold or concentration of beryllium below which acute disease does not occur, despite widely varying incidence rates within single plants themselves. This position is much less tenable, however, for the chronic disease. A continuing controversial subject is the so-called "neighborhood case," a name applied to cases of beryllium disease where exposures have apparently been indirect. This was first described in 1938 by Gelman,[46] and later by Hardy[47] and Eisenbud.[48] As of 1960, 47 such cases had been reported in the literature. Paraphrasing Tepper, Hardy, and Chamberlin,[37] this number unquestionably represents only an unknown fraction of such cases, for the diagnosis cannot be made when it is not suspected as the possible cause of diseases, and many patients with minimal symptoms have almost certainly been overlooked.

Medical Control

It is obvious from what has been said regarding the epidemiology and treatment of the disease that the most important means of control lies in its prevention. No known method of treatment restores the good health the patient might have had if he had never developed the disease. The first line of defense rests primarily with the industrial hygienist and engineer. Physicians function as a secondary defense.

The application of well-established health hygiene principles has resulted in a negligible number of new cases of beryllium disease in recent years. These practices have included adequate sampling and analysis of air, properly engineered equipment to collect dust and fumes, good

housekeeping, and instruction of exposed personnel in personal hygiene. However, it would be presumptuous, even today, to claim that the disease has been conquered by preventive means. It is also presumptuous to state that the criteria of safety (in terms of the permissible limits of the concentration of beryllium and its compounds in air) have been established on a firm basis. It does appear that the hygienic standards first set forth by the Advisory Committee to the USAEC in 1949, and later accepted by the American Conference of Governmental Industrial Hygienists and the American Industrial Hygiene Association, are adequate. Despite periodic reconsiderations of available information by these groups, the original recommendations have not been changed. These well-known standards are: (1) The inplant atmospheric concentration of beryllium should not exceed 2 $\mu g/cm^3$, as an average, over an 8-hr day. (2) Even though the daily average might be within the limits of recommendation (1), no person should be exposed to concentrations greater than 25 $\mu g/cm^3$ for any period of time, however short. (3) In the neighborhood of a plant handling beryllium compounds, the average monthly concentrations at the breathing-zone level should not exceed 0.01 $\mu g/cm^3$.

No provisions were made in these recommendations with respect to either the physical or chemical nature of the beryllium compound involved. To date, no cases of either acute or chronic beryllium disease have been reported in connection with any plant operation in which these standards have been maintained. There is insufficient data at present to establish alternative limits, so, although these standards are not precise criteria and their scientific basis is weak, they are still the best available guides in current technological practice.

Medical control, although secondary, should not be ignored. At the time of the pre-employment examination, the physician should examine the prospective beryllium worker critically with respect to the risk of that person developing beryllium disease. People with inadequate intelligence or regard for the health of themselves or others should not be permitted to handle potentially dangerous materials; persons with skin disease that might prove unduly sensitive to soluble beryllium salts should not be exposed; and persons with chronic respiratory tract disease which, in theory, at least, might be associated with poor or damaged defense mechanisms or dimished pulmonary reserve might also be rejected. People with abnormal chest x-rays or with a history of repeated respiratory infections, with significant past exposures to potentially injurious industrial dusts, such as silica, with organic heart disease, or with tuberculosis should also be screened carefully. Certainly there is no evidence that these disorders will influence the development of beryllium disease; however, they may be associated with changes in health which are difficult to distinguish from chronic beryllium disease.

Periodic examinations of beryllium handlers are a part of a sound health-maintenance policy, as they would be in any industry or in private medical practice. Particular inquiry should be made regarding respiratory symptoms. If a known over-exposure occurs, the affected person should be examined carefully and at more frequent intervals, with particular reference to the possibility of his contracting the acute disease. The objective of both the periodic examinations and exposure examinations is detection of minimal disease, and certainly is not to be regarded as a form of protection. The checkups are a means of picking

up a potential disease process as early as possible. There is no evidence to date that routine examinations of the blood and urine or vital capacity are helpful in the control of beryllium disease. The beryllium patch test should be avoided as a routine test, since it may be a sensitizer.

Summary

General agreement exists today that beryllium and its compounds are directly responsible for toxic effects among exposed persons. The clinical responses are not unique and are similar to those found with other diseases. There is a wide spectrum in the clinical picture, with no clearcut differences between acute and chronic effects. The most typical, serious, acute toxic effects involve the lungs and result in a chemical pneumonia; there is a definite dose-effect relationship, and recovery is the rule. The chronic disease, like the acute disease, most significantly affects the lungs, but unlike the former is a slowly progressive disabling disease for which a dose-effect relationship has never been adequately determined. It is, moreover, a systemic disease and is characterized by the fact it is frequently separated by a period of years from the time of exposure, is prolonged in duration with little evidence for cure, and is commonly progressive in severity despite cessation of exposure. All industrial beryllium compounds, including the metal (except for beryl, the natural ore), have been reported as having caused some manifestation of the disease. Patch tests, though potentially useful in diagnosis and a means of screening, have considerable limitations and may cause exacerbation of beryllium disease in a sensitive person;

they are felt by most authorities today to be of little value. Control of the disease is based on application of good industrial hygiene principles and prevention and not on treatment. Steroids have been found to be useful in favorably affecting the course of the disease but not in effecting a cure. Chelating agents, although showing promise in experimental applications, have not proved to be of any practical value. (Gino Zanolli, M.D.)

JCAE HEARINGS ON RADIATION STANDARDS, INCLUDING FALLOUT

The Subcommittee on Research, Development, and Radiation of the Joint Committee on Atomic Energy (JCAE) held public hearings in Washington, D. C. on June 4-7, 1962 on "Radiation Standards, including Fallout."[49]

> "The subject matter of the hearings proved to be particularly timely because of the mounting public and congressional concern over the extent of the hazards from fallout due to renewed testing of nuclear weapons in the atmosphere, first by the Soviet Union in the fall of 1961, and later by the United States in the spring of 1962.... A major purpose of these hearings, as in the case of those in the past, was to obtain a better perspective on all aspects of radiation hazards and of the standards which have been established as a guide to protection against these hazards... . Of particular interest to the committee were the role and responsibilities of the Federal Radiation Council, which remained unclear at the conclusion of the 1960 radiation standards hearing.
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> "Several pertinent documents became available for study prior to or during the hearings,... [including reports from the Federal Radiation Council,[50] the Public Health Service[51] and the Atomic Energy Commission.[52]] The Federal Radiation Council report discussed the possibilities of adverse health effects from tests conducted up to the end of 1961. It also reviewed available data, derived from experiments with mice, comparing the effects of low- and high-level radiation exposure on the number of mutations produced. The National Advisory Committee on Radiation report for the Public Health Service examined the needs for countermeasures and surveillance to combat environmental radiocontamination. The proceedings of the Conference on Radioactive Fallout From Nuclear Weapons Tests included a number of technical papers by active workers in the AEC fallout research program and associated research efforts.
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> "To bring the subject of radiation standards and fallout up to date, the committee considered certain major points as listed in the summary analyses of previous hearings. These included in the field of radiation protection standards:*
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> (a) The nature and development of radiation protection standards and criteria.

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*See 1960 summary-analysis of the hearings, Radiation Protection Criteria and Standards: Their Basis and Use, p. 2.

(b) Activities of Federal, State, and private agencies and groups concerned with radiation protection, criteria and standards.

(c) Concepts underlying radiation protection criteria and standards: Biological concepts; social and economic concepts; concepts of risk; and concepts of practical application, particularly in dealing with operational applications related to new sources such as fallout.

(d) Organizational, administrative, and legislative problems. In the field of fallout, they included:*

[1] Origin and distribution of fallout.
[2] Uptake of radionuclides through food chains.
[3] Biological effects of radiation.
[4] Population exposures from past tests.
[5] Population exposures from future tests."

The hearings lasted four days, during which time many personages from government, scientific institutions, and business presented testimony on the various topics considered. All this testimony has been incorporated into the record of the hearings, copies of which are available. The record of the hearings is voluminous, and it is hardly possible to summarize fairly the various positions of the different individuals and organizations on the issues discussed. Hence the reader is encouraged to refer to the Committee report for background information, specific data, and details; however, some of the apparent highlights and trends presented in the testimony are discussed here. All quotations below are taken from the report on the hearings.[49]

Radiation Protection Standards

According to the testimony there has been no basic change in industrial application of radiation protection standards since 1960, although

*See 1959 summary-analysis of the hearings, Fallout From Nuclear Weapons Tests, p. 2.

the application of these standards has been clarified by Federal Radiation Council Report No. 2.[53,54] The average annual exposure of industrial workers at atomic energy facilities in 1960 and 1961 was ~1/3 rem per year, which is to be compared with the permissible 5 rem per year, whereas the additional dose to persons living in the vicinity of nuclear plants was estimated to be less than 1 mr per year, which is to be compared with the 500 mr in 30 years suggested by the ICRP.[55] Recent data also indicate that the average medical exposure to individuals in the population is one-fourth to one-half the 0.1 rem per year previously estimated.

Consideration was given to the establishment of additional criteria on several aspects relating to radiation exposure including: (1) exposure of patients for medical purpose, (2) radiation-protection and measurement devices, (3) application of limits established for peaceful application to situations involving fallout from nuclear weapons tests. Additional criteria were not, however, recommended in any of these cases, since it was felt not necessary in the first instance, premature in the second, and no progress had been made in studies of the last.

> "Based on an analysis of the current status of radiation protection standards as reflected in our various hearings, and considering the wide variety of radiation producing sources, including fallout, the subcommittee believes that the following identifies the main problems which exist in this area:
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> "1. The major issues in the field of radiation protection standards involved their application to the population from an increasing variety of sources, particularly those leading to radioactivity in the environment; for example, fallout, waste disposal, space application, and so forth. Although problems remain in the area of standards for occupational exposure, the major deficiencies appear to lie in population exposure standards. [This point has been raised repeatedly in reactor siting consideration.[56,57]]

"2. Some fundamental questions related to exposures of the population appear to be unanswered; for example: What purposes are the radiation standards supposed to achieve? Just what are radiation protection standards trying to do? ... Under present philosophies of radiation protection is there some fixed value of radiation exposure from all sources which cannot be exceeded without undue risk to health? As a matter of national policy or philosophy, should radiation protection standards be applied to all programs of the Government, including those required by national security and new applications under development? Is it either necessary or desirable for population standards to have a fixed numerical relation to occupational standards? ...

"3. Effective leadership for developing radiation protection criteria and standards to cover new operational problems such as fallout is not adequately exercised by any government agency or group at present. ...

"4. There continues to be no effective means by which social and economic factors can in fact be applied in concert with other major considerations in the development of radiation standards for application to fallout unless this can be achieved through the staff or by other mechanisms of the FRC. ... The subcommittee believes that there is an increasing urgency for Government action in this area and that the Federal Radiation Council is neither adequately staffed nor organized for this purpose."

Problems Concerning Fallout

Most of the testimony at the hearing pertained to fallout, which was subjected to increased attention because of the resumption of nuclear testing by both the United States and the USSR. The problem of greatest concern at these hearings was I^{131}, which has an 8-d half life. People are exposed to fallout iodine principally through fresh-milk consumption, from whence it concentrates in the thyroid and is presumed to cause thyroid cancer. However, no consistent relationship between the concentration of fission products in air and I^{131} in milk has not yet been discovered. "Cumulative levels of iodine 131 in milk are, in some areas, approaching or possibly exceeding the acceptable levels recommended by the Federal

Radiation Council for environmental sources generated by peaceful applications. What levels of this nuclide can be tolerated for exposure of a population from fallout have not been established."

Methods were presented for the prediction of probable exposures from stratospheric, world-wide fallout from weapons testing under three conditions: (1) at low altitudes in polar latitudes, (2) at low altitudes in equatorial latitudes, and (3) at high altitudes in any latitude. The distribution of tropospheric fallout was, however, considered to be unpredictable.

> "A new concept, called the dose commitment concept, was introduced during the hearings to explain the significance of fallout. The fundamental hypothesis of this concept is that the risk of biological damage to a large population is proportional to the average radiation exposure level produced by radioactive contamination in the environment [regardless of when the exposure is received]. Thus, it is not necessary to specify the biological effect, because the same effect should be produced by an equivalent exposure to radiation from natural sources.* Previous comparisons have been based on a 30-year exposure for genetic effects and a 70-year exposure for somatic effects compared to natural background during 30 and 70 years, respectively.
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> "The major problem areas in this field appear to be:
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> "1. While our ability to predict the general pattern of global fallout appears reasonably well developed for certain patterns of testing, it is quite clear that many local situations exist for which prediction is not well worked out.
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> "2. There are still major gaps in basic information on what biological effects relate to what levels and rates of radiation exposure. These gaps can be expected to remain for some time.
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> "3. Development of an approach toward evaluating transitory high dietary or air levels of one or more short-lived nuclides is needed. The committee believes that recurring periods of such levels will be experienced from time to time in the future.

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*At the same time, the validity of this assumption depends ultimately on the validity of the linear hypothesis for dose-effect relationships.

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II. ACCIDENT ANALYSIS

INTEGRITY OF REACTOR FUELS

The integrity of reactor fuel elements plays a major rôle in the economy, as well as the safety, of nuclear reactor operation. The consequences of fuel element failure may range from contamination of the reactor coolant stream to the release of significant amounts of activity to the atmosphere, and, yet, the longer a fuel element can operate without failure, the greater is the benefit derived from a given expense for fuel fabrication and recovery and the less are the expense and downtime for refueling and decontamination. Thus extensive irradiation testing and postirradiation examination of nuclear fuels are carried out to guide the operation of reactors, to evaluate the integrity of new fuels, and to aid in the development of fuels with higher burnup capabilities and less severe consequences of failure.

Fuels subjected to postirradiation evaluation take many forms. These include fuel elements removed from reactors after failure or after periods of normal operation, fuel elements or prototypes irradiated to destruction in reactor transients and in steady-state reactor operation, and fuel assemblies and clad and unclad fuel specimens irradiated in operating reactors under conditions simulating, to various degrees, anticipated conditions of use. The fuels examined include some of the types already established by reactor use, modifications of these made to meet demands for longer life and higher performance, and new materials for future reactors. Previous reviews[1,2] treated several aspects of this topic, and the results of recent tests of various types of fuel are presented here.

Metallic-Uranium-Based Fuels

Much of the current information on metallic-uranium-based fuels relates to elements clad to meet the conditions in power reactors and to uranium alloyed to improve its dimensional stability. Although most of the work reviewed here is based in U. S. technology, some Canadian investigations are discussed.

Fuel Element Failure in the SRE. A multiple fuel-element failure in the Sodium Reactor Experiment (SRE) in July 1959 required extensive repair and modification to the reactor system.[3,4] Examination of the failed fuel elements revealed two modes of failure.[5,6] In one, thermal cycling and radiation damage caused swelling of the uranium fuel that was sufficient to rupture the cladding. In the other, interdiffusion of uranium and the stainless steel components formed low-melting alloys that penetrated the cladding. The occurrence of either of these mechanisms required overheating of the fuel element, which occurred because the coolant channel was blocked by reaction products of tetralin that had leaked into the sodium system, probably from the tetralin-cooled frozen-metal seals in the sodium pumps.

The reactor has now been restored to operation following several modifications to prevent recurrence of the damage.[5] For example, the tetralin coolant has been eliminated from the reactor system to avoid the source of channel blockage, and, the fuel elements have been made smaller and structurally stronger to facilitate removal in the event of failure. Further, the uranium fuel has been replaced with a thorium-base alloy that is less susceptible to either of the types of damage

found in the defective elements because of its isotropic structure and the higher melting point of the thorium-iron eutectic (1580°F compared with 1340°F for uranium-iron eutectic).

Capsule irradiations of uranium samples to exposures of 2260 Mwd/MT, simulating SRE conditions, showed[7] volume increases up to 15.1% for alpha-rolled uranium and 13.1% for cast uranium. Also, examination of fuel elements removed from the SRE after an exposure of 850 Mwd/MT, that is, before the failure of several fuel elements, had revealed unsatisfactory dimensional stability of the unalloyed uranium.[4] In fuel elements irradiated to about 1000 Mwd/MT, swelling and warping of the uranium slugs led to displacement of the NaK thermal bond and stretching of the stainless steel can. Of some experimental fuels tested in the reactor, an alloy containing 2% zirconium was not much better than unalloyed uranium, but uranium–1.5% molybdenum and thorium–5.4% uranium alloys showed excellent dimensional stability.

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Uranium Alloy Fuels. Two of a series of eight stainless steel capsules containing uranium-molybdenum specimens failed during irradiation in the Materials Testing Reactor.[10] The failures were found to be due to stress-corrosion cracking of the outer capsules, which had been sensitized by a preirradiation heat treatment. The cracks permitted cooling water to leak slowly into the helium annulus about an inner stainless steel capsule. In one of the capsules, a leak probably developed in the inner capsule and permitted steam to consume the NaK thermal bond and the fuel alloy. The other inner capsule was still intact, and the contained fuel alloy showed only slight dimensional changes (1.5% volume increase).

Fast-reactor fuel pins with various types of cladding have been tested to failure in reactor transients that provided a rapid temperature rise by nuclear heating.[11] Most of the experiments were with EBR-II Mark I fuel alloy, which is uranium with additions of 2.5% molybdenum, 1.5% ruthenium, 0.5% palladium, 0.3% rhodium, and 0.2% zirconium. Although the actual reactor fuel contains uranium of 49% enrichment, natural uranium and uranium enriched to 3% U^{235} were used in these experiments to obtain suitable heating rates. Stainless steel-clad EBR-II Mark I elements failed upon reaching cladding temperatures above 960°C during brief transients, and they failed violently at temperatures above

1015°C. Failures also occurred at temperatures down to 600°C in transients in which the peak temperature was maintained for several seconds. The failures resulted from dissolution of the stainless steel by eutectic formation with the uranium. Much higher temperatures were required to cause failure of similar elements clad in niobium or tantalum; these survived brief transients up to 1400°C. Enrico Fermi fast breeder reactor core I elements, which are zirconium-clad uranium–10% molybdenum alloy enriched to 10% U^{235}, showed damage but no failure below 1300°C.

Stainless steel-clad elements fueled with both cast and wrought uranium–2% zirconium alloy were irradiated in the EBR-1 to burnups of 0.08 to 0.189 at. %.[12] Dimensional stability was better for the cast material than for the wrought fuel slugs and was adversely affected by annealing of the wrought material. On the other hand, Block[13] has reported that uranium enriched to 10% U^{235} and alloyed with 1 or 2% iron showed serious distortion upon irradiation to a burnup of 0.1% of the uranium.

Thorium-Based Uranium Fuels

The highly anisotropic crystal structure of uranium gets much of the blame for the distortion of uranium that occurs under irradiation. Thorium, on the other hand, has a cubic structure that should resist such distortion. Also, thorium does not undergo phase transformation in the temperature range normally or accidentally encountered in metal-fueled reactors. As indicated above, a thorium-base fuel showed excellent dimensional stability in experimental fuel elements in the SRE, and this fuel is now being used for the full core loading in the reactor.

Kittel and Horak[14] have examined irradiated thorium alloys containing up to 31% uranium. Alloys containing 20% uranium or less showed

excellent dimensional stability. Severe distortion occurred in alloys of higher uranium content, consistent with the appearance in these alloys of a continuous network with the alpha-uranium structure.

Oxide-Based Fuels

Uranium dioxide has gained widespread use as a nuclear fuel because of its stability under irradiation and high temperatures and in the presence of water. The effects of irradiation on the physical properties of this oxide have been reviewed by Roake.[15] The factors influencing the integrity of UO_2 fuel elements in water-cooled reactors were reviewed previously,[2] and irradiation testing of UO_2 fuels has continued. Also, extensive testing is under way of UO_2 diluted with other ceramic materials. The most prominent diluent is the fertile ThO_2, but other diluents have been examined for improved thermal properties.

Uranium Dioxide Irradiation Tests. Gray and Mrazik[16] have examined a pressurized water reactor (PWR) type of Zircaloy-2-clad UO_2 fuel rod that failed after 1.93 at. % burnup in a Materials Testing Reactor (MTR) test. It is believed that a cladding defect developed during the test and admitted water. The resulting superheated steam corroded the inside of the cladding and produced hydrogen, which embrittled the cladding and led to cracking. A possible contributing factor in the failure was that a short time previously the test equipment had been disassembled for repairs and reassembled in such a way as to expose this rod to a higher flux than intended. Other rods in the test assembly survived a much higher exposure.

Examination of similar PWR elements exposed under more normal conditions has been reported by Bleiberg[17] and by Center and his co-workers.[18]

In general, no serious irradiation effects were observed. The latter report[18] cites an instance of a blanket rod performing satisfactorily at 2.22×10^{20} fissions/cm^3 with a heat flux of 98,700 Btu/hr·ft^2 despite a 0.004-in.-diam penetration of the 0.027-in. cladding. The penetration apparently developed early in the exposure as a result of cladding defects introduced in fabrication and not as a result of irradiation.

The catastrophic failures that sometimes occur in UO_2 elements with defective Zircaloy cladding have been attributed by Notley and his coworkers[19,20] to fluoride impurities in the UO_2. They found failures in fuel elements with 0.025-in. defects in the cladding and 500 ppm fluoride in the uranium dioxide. Similar fuel elements made from UO_2 containing only 50 ppm fluoride performed satisfactorily under the same conditions. Although the investigators did not propose a mechanism for the failures, it appears that fluoride ions released from the oxide accelerated corrosion and hydriding of the cladding.

Notley[21] has observed the dependence on diametral clearance and contact temperature of the relative motion during irradiation of Zircaloy-2 cladding and stacked UO_2 pellets. Parry[22] has examined two Zircaloy-2-clad UO_2 fuel elements, one of which failed, after irradiation in the NRX for 16.2 days at a heat flux of 78 w/cm^2. In both elements the uranium oxide reacted with the Zircaloy end caps; there was evidence that the interface reached a temperature of 1400°C. The cause of the failure was not determined, but heavy hydriding of the cladding indicated that water had entered the fuel element.

A maximum burnup of 6100 Mwd/MT in a pressurized-water loop at 240°C was survived with negligible damage by a prototype three-rod element

comprising UO_2 pellets clad by collapsed A-288 aluminum-alloy tubing.[23] Cole[24] has found little dimensional change upon irradiation up to 3390 Mwd/MT of swaged rods of stainless steel-clad UO_{2+x}.

Irradiation testing of prototype fuel elements for the Experimental Gas-Cooled Reactor (EGCR) has been reported.[25,26] The test element, identical with the actual element except in length, is a stack of hollow, cylindrical, dense UO_2 pellets between two magnesia spacers in a type 304 stainless steel tube. Since it was feared that fracturing of the pellets might lead to fuel chips dropping through the central void, alternate elements with porous beryllia plugs in the center and elements with solid pellets were simultaneously tested. The elements were exposed in the Oak Ridge Research Reactor (ORR) and the ETR in 300-psi helium with cladding temperatures of 1300 or 1600°F. Unimportant dimensional changes were observed in irradiations up to 2400 Mwd per MT of UO_2. At the higher temperature, ridges containing voids and sigma phase were formed in the stainless steel at the fuel-pellet interfaces that could lead to failure after longer exposures. The pellets showed considerable cracking, but the cracks were not of a type that would impair the integrity of the fuel elements. Some bowing was observed in elements with solid pellets or beryllia cores.

Irradiation Behavior of Thorium Oxide Fuels. In pellets of ThO_2–10% UO_2 (fully enriched)–1% CaF_2 irradiated to about 20,000 Mwd/MT, the resistance to fracture was found[27] to increase markedly with increasing open porosity but to be unrelated to closed porosity. Pellets of ThO_2 containing up to 25.61% UO_2, lead-bonded to X8001 aluminum cladding, have also been irradiated to high burnups.[28] A specimen irradiated to

23,400 Mwd/MT increased 1.7% in volume. The two specimens with the highest uranium content failed after 27,000 Mwd/MT at a heat flux of 7×10^5 Btu/hr·ft^2.

The incorporation of molybdenum fibers in ThO_2-UO_2 fuel pellets markedly improved stability to radiation.[29] The onset of damage occurred after considerably longer exposures than in similar unreinforced pellets. Stability of the pellets was also increased by the use of lead rather than helium as a heat-transfer bond to the Zircaloy-2 jacket. The metal fibers improved thermal conductivity and resistance to thermal shock. Similar use of niobium fibers was of no value because the niobium reacted with the oxides.

Insignificant dimensional changes upon irradiation in the Experimental Boiling Water Reactor (EBWR) for 1000 Mwd/MT were found[30] for an Elk River fuel assembly consisting of ThO_2–3% UO_2 pellets in type 304 stainless steel tubing. The element was exposed to a maximum flux of 8.0×10^{12} neutrons/cm^2·sec and the maximum surface heat flux was 85,000 Btu/hr·ft^2.

Fuels with Other Ceramic Diluents. Considerable fracture occurred in pellets of ZrO_2–10% UO_2–9% CaO irradiated bare in NaK to burnups ranging from 11,300 to 17,200 Mwd/MT.[31] Fuel elements of UO_2 and UO_2 diluted with BeO and clad in Hastelloy-X have been tested for use in mobile high-temperature air-cooled power reactors (such as the ML-1).[32] In capsule irradiations in the MTR to 13% U^{235} burnup, negligible damage occurred in BeO–70% UO_2 (18 mole % UO_2), but one specimen containing BeO–80% UO_2 (27 mole % UO_2) swelled enough to cause cladding failure. A contributing cause of the failure was a decrease in thermal conductivity of the fuel

because of a change from a BeO matrix to a UO_2 matrix during irradiation. In a fuel element sustaining burnup varying with position from 5800 to 24,300 Mwd/MT (0.69 to 2.9% of the U^{235}) in a nitrogen-cooled loop in the General Electric Test Reactor (GETR), both UO_2 and the UO_2—BeO mixture showed good dimensional stability, although the undiluted UO_2 pellets showed more cracking. Although the Hastelloy-X cladding did not fail during irradiation, it became so embrittled by a grain-boundary precipitate that two of the 12 fuel tubes broke during postirradiation examination.

Alumina-coated, fully enriched, UO_2 particles dispersed in graphite have been irradiation-tested by Burian and his co-workers.[33] After burnups of 4.7 and 5.6% of the U^{235} in the Battelle Research Reactor at 510 and 730°C, little irradiation damage was noted in the graphite. The alumina coatings softened and about 8% of them cracked.

Uranium Carbide Fuels

Sinizer and his co-workers[34] have irradiated uranium carbide thermally bonded to stainless steel capsules with sodium or NaK. For burnups ranging from 2000 to 25,000 Mwd per metric ton of uranium at central fuel temperatures from 800 to 1900°F, the growth in diameter of the carbide fuel pieces could be empirically correlated by the relationship

$$R_D = 0.6 + 0.77 \left(\frac{T_F - 541}{1000}\right)^5,$$

where R_D is the ratio of the diameter increase (in percent) to the burnup (in atom percent) and T_F is the central temperature (in degrees Fahrenheit). Radial and transverse cracks were observed in the slugs but were

not believed to be detrimental. Hyperstoichiometric carbides (>4.8% C) transferred carbon through the molten metal bond to the stainless steel capsule and embrittled the steel.

Osborne and Morgan[35] have irradiated pellets of UC clad in tantalum to burnups up to 4610 Mwd per metric ton of uranium at initial central temperatures up to 2290°F. In all cases, reaction was observed between the carbide and the tantalum. Cladding failures occurred in two of the six capsules.

Dispersions of fully enriched UC_2 in graphite spheres coated with either pyrolytic carbon or siliconized silicon carbide have also been irradiated.[33] The carbide-coated spheres attaining 4 and 6.5% U^{235} burnup cracked badly and the one with the higher dose fell apart. A pyrolytic-clad sphere showed line cracks in the cladding at a burnup of 5%, whereas another was in good condition, with no leakage from the cladding, at a burnup of 3%.

Aluminum-Based Fuels

Aluminum-based fuels have long been used in research reactors because of the ease of fabrication and reprocessing, despite their restricted heat rating. Continued development, including irradiation testing, of aluminum-based fuel elements is aimed at achieving good performance at higher burnups, higher fluxes, and higher temperatures while retaining the ease of fabrication and reprocessing of these fuels. A serious failure of an aluminum fuel element in the Westinghouse Testing Reactor (WTR), reviewed in detail previously,[36] and recent developments in connection with other aluminum-based fuels are described here.

Alloy and Cermet Fuels for Research Reactors. Francis and his co-workers[37-39] have examined fuel plates with aluminum cores containing up to 50% uranium or 65% UO_2, irradiated to burnups of up to 50% of the uranium. The claddings tested have been 1100-, 5052-, 6061-, X8000-, and APM-series aluminum alloys. Leitten and his co-workers[40,41] have similarly studied dispersions in aluminum of 64% UAl_3, 63% U_3O_8, and 60% UC_2. The changes observed [36,37] are, in general, modest dimensional changes that are seldom greater than 3%, loss in ductility, and increases in hardness and strength. Dispersed uranium oxides react with the aluminum without affecting the integrity of the fuel. A full-scale fuel element containing the 63% U_3O_8 cermet, with the uranium enriched to 20% U^{235}, showed excellent dimensional stability upon irradiation in the ORR to an average U^{235} burnup of 41% (57% peak).[40,41]

Sample alloy fuel plates have performed satisfactorily[37] with two types of duplex cladding. One irradiated to a burnup of 38% had an inner 12 mils of APM M257 aluminum for mechanical strength and an outer 3 mils of X8001 aluminum for good corrosion resistance. A burnup of 28% was achieved in an alloy fuel with an inner 1 mil of 2% boron-aluminum alloy for burnable poison and an outer 14-mil aluminum cladding.

A gamma-scanning technique has made possible determination of burnup contours in fuel plates from the MTR and ETR.[37] The burnup in MTR plates peaks about 2 in. below the center line. In the ETR, burnup is more symmetrical, peaking very close to the center line. In both reactors, the burnup is greater at the plate edges than at the axis. These results show less flux asymmetry than the customary flux-wire measurements. The same fuel plates showed negligible defects in ultrasonic inspections.

Irradiation of Prototype SL-1 Fuel Plates. A prototype SL-1 fuel plate failed because of local corrosion of the X8001 aluminum cladding after 58% burnup of the U^{235} in a 420°F circulating-water loop in the MTR.[42,43] The fuel core, an alloy of aluminum with 17.5% uranium, 2% nickel, and 0.5% iron, was sufficiently resistant to attack by the water that the radioactivity level in the loop did not require shutdown until two days after the failure. The penetration of the cladding was attributed to the small ratio of aluminum surface to loop water volume, aggravated by a period of operation at a high pH (10) and by overheating because of a scale deposit. A similar plate irradiated to 45% U^{235} burnup showed only modest corrosion.

Fueled-Glass Dispersions. Dispersion plates of fueled glass in aluminum have been tested[44,45] in the MTR to 17 to 24% burnup of the uranium. The fuel cores were compacts of equal volumes of aluminum and glass fibers that contained 50% fully enriched U_3O_8. The irradiation caused no changes in microstructure, no visible fission-fragment damage, and no reaction of the glass with aluminum. Changes in dimensions and tensile properties were in the normal range for uranium-aluminum dispersion fuels.

Stainless Steel-Dispersion Fuels

Weir[46] has reviewed the irradiation behavior of dispersions of UO_2 in stainless steel and has theoretically analyzed the mechanism of damage. He concludes that a high burnup potential is favored by decreasing the volume fraction of the UO_2, increasing the UO_2 particle size, and decreasing the temperature.

Freas and his co-workers[47] have examined type 318 stainless steel-clad dispersions of 30% fully enriched UO_2 in type 318 stainless steel after irradiation in the MTR at estimated central temperatures ranging from 800 to 1600°F and burnups ranging from 10 to 15% of the U^{235}. Both spherical and hydrothermal uranium oxide was used, but all the irradiated specimens were free of swelling, cracks, and cladding defects. Increases in thickness were all under 3%, and decreases in density were under 1%.

Paprocki and his co-workers[48] have examined type 347 stainless steel-clad dispersions of 26% fully enriched UO_2 in type 347 stainless steel after irradiation in the MTR and ETR. Three specimens failed because of inadvertent local overheating. Otherwise, the performance was satisfactory. The maximum density decrease was 3.1%, and cracks introduced in the oxide by fabrication were healed by irradiation. No adverse effects resulted from inclusions of burnable poison in the form of B^{10} in the cladding or ZrB_2 or NbB_2 dispersed in the fuel core.

Heffner[49] has studied damage to stainless steel-dispersion fuels during progressively more rapid transients in Special Power Excursion Reactor (SPERT I and III) tests. As the periods of the transients decreased to about 14 msec, rippling of the fuel plates was observed. At shorter periods, the plates bowed and, in the regions of maximum neutron flux, developed blisters. After repeated transients or periods as short as 5 msec, these blisters burst and exposed the fuel. The blisters did not result from weakening of the cladding but, rather, from voids formed beneath the cladding and along the UO_2 stringers.

Plutonium-Containing Fuels

Specimens of PuC and UC-201 PuC were irradiated in the EBR-I to burnups from 500 to 770 Mwd/MT at central temperatures of 370 to 650°C.[50] The mixed carbide specimens decreased in density, but the dimensional changes were unimportant. The specimens with more carbon atoms than plutonium and uranium atoms cracked. Most of the plutonium carbide specimens fractured considerably.

Fuel alloys thorium-5% plutonium and thorium-10% plutonium showed adequate dimensional stability upon irradiation at 450°C to 1.9 and 2.6 at. % burnup, respectively.[51] The alloys zirconium-5% plutonium and zirconium-7% plutonium were badly distorted. Fuel rods of plutonium-1% aluminum, NaK bonded in Zircaloy-2 tubes, similar to Mark IV fuel elements for the EBR-I, showed unimportant dimensional changes upon irradiation at 385°C in the EBR-I to a burnup of 0.1 at. %.[52]

Horak and his co-workers[53] have reported on the irradiation behavior of uranium-based plutonium-containing fuel slugs, which are prototypes of fuel for the second core of the EBR-II. The principal alloy investigated was 70% uranium, 20% plutonium, and 10% fissium, the latter being 4.6% zirconium, 25.9% molybdenum, 39.8% ruthenium, 6.5% rhodium, and 23.2% palladium, a mixture representative of the fission products accumulated in recycle fuel after repeated pyrometallurgical reprocessing. The fuel specimens remained in good condition to better than 1% burnup at central temperatures less than 370°C. Swelling and warping became progressively worse at higher irradiation temperatures. Fuel stability was improved by lowering the plutonium content to 10% but impaired by increasing the

fissium content to 15%. The alloy uranium–20% plutonium–5% molybdenum was less stable than the reference alloy.

Bailey[54] has reported irradiation tests of several Zircaloy-clad plutonium fuels representative of fuels intended or considered for the Plutonium Recycle Test Reactor (PRTR). Capsule irradiations at the MTR included aluminum-based alloys containing 1.65 to 20% plutonium and either 0 or 12% silicon. Also tested were UO_2–PuO_2 mixtures containing 0.0250 to 7.45 mole % PuO_2 and having either low (~65% of theoretical) or high (~90%) density. Four- and seven-rod cluster alloy fuel elements were tested in a Hanford in-pile loop.

Exposures of the capsules ranged from 1.4×10^{19} to 10.5×10^{19} fissions/cm^3 at 46°C. In general, the performance was satisfactory. The only failure was a high-density oxide sample exposed to a flux eleven times greater than intended. The alloy fuels without silicon showed the best dimensional stability. The high-density oxides showed better fission-gas retention than those of low density. Grain growth was observed in one high-density specimen, and central void formation was noted in two specimens of low density.

The four-rod cluster element contained 24 g of plutonium and consisted of 8-in. rods of aluminum–8% plutonium and aluminum–8% plutonium–12% silicon in Zircaloy-3 tubing with Zircaloy-2 end caps. Exposure to 3.0×10^{20} neutrons/cm^2 at 185 to 210°C in pressurized water of pH 6 led to measurable but acceptably low dimensional changes. The two fuel slugs comprising one of the rods diffusion bonded during the irradiation.

The seven-rod cluster contained 20 g of plutonium and was fueled with 18.3-in. rods of aluminum–1.8% plutonium clad in Zircaloy-2. Acceptably

low dimensional changes were observed after an exposure of 4.7×10^{20} neutrons/cm^2 in water (pH 10) at 230 to 310°C.

Conclusions

It is clear that dimensional instability severely limits uranium metal to relatively low burnups and temperatures. When clad with stainless steel, overheated uranium and the steel form low-melting alloys that penetrate the cladding. Alloying additions to uranium, in particular molybdenum, improve its performance, but the burnup and temperature limitations are still below those of other fuel materials. Plutonium-base fuel alloys are similarly limited.

When a concentrated fuel is needed, as in a low-enrichment uranium reactor or a fast reactor, performance at high temperatures and high burnups requires a ceramic fuel. Uranium dioxide has shown excellent performance to very high burnups. The stability of the oxide toward water has enabled oxide fuel elements to perform satisfactorily in water-cooled reactors, despite defects in the cladding, when the oxide was free of fluoride and the elements were free of defects through which water could enter but from which steam could not rapidly escape. Uranium monocarbide is another promising fuel, although there is at present inadequate experience with its use. Although ceramic fuels crack during irradiation, it is not expected that cracks will impair their performance.

When dilution of the reactor fuel is permissible, many satisfactory systems can be considered. Uranium and plutonium may be alloyed with several diluent metals, and ceramic compounds of these elements may be dispersed in both metal and ceramic diluents.

Aluminum-based fuels have had long use in research reactors. Use of new alloys with improved strength and corrosion resistance has raised the attainable operating temperatures and heat fluxes for these fuels. Increased uranium concentrations can be obtained by alloying or powder-metallurgical techniques that avoid UAl_4 formation or by dispersion of a ceramic, such as U_3O_8. Dispersed oxides react with aluminum under irradiation, but the reaction apprently does not impair the integrity of the fuel element.

When gains in fuel economy and prolonged reactivity from conversion are desired, thorium-based fuels may be used. Uranium and plutonium have both been incorporated in thorium fuels and tested successfully at temperatures and burnups greater than those at which uranium-based fuels show serious distortion. Poor performance has been observed, however, for thorium fuels containing greater than 20% uranium.

Special-purpose water-cooled power reactors can use fuels based on zirconium or stainless steel. While this review has not uncovered recent information on zirconium-alloy fuels, their long and expanding use in Naval reactors attests to their integrity. The cermet fuels of uranium oxide in stainless steel have had less extensive use, but encouraging information on their stability and limitations is now becoming available. Preliminary information on zirconium-base plutonium fuels is discouraging.

Increasing interest is being shown in all-ceramic dispersion fuels. Solid solutions of uranium dioxide in thorium oxide show the same high performance as the pure uranium oxide fuels, and the thermal properties of this material can be improved by incorporation of molybdenum fibers. For high-temperature systems, dispersions of UO_2 in ceramics such as

beryllia, zirconia, and graphite and dispersions of UC_2 in graphite are being examined. Although experience with these fuels is not yet extensive, several appear promising. (S. Peterson)

STRESS-CORROSION CRACKING OF STAINLESS STEEL

Of the types of corrosion that can plague reactor systems, none is more insidious or damaging than stress-corrosion cracking. This type of attack frequently occurs under conditions where the metal or alloy in question is resistant to general attack. Stress-corrosion cracking results from the combined action of a tensile stress, either applied or residual, and a specific corrodent on a metal or alloy; in the absence of either, failure does not occur by this mechanism. Failure manifests itself as fine, hair-line cracks, which can be either inter- or transcrystalline, depending on the corrodent and the metallurgical condition of the alloy. The time necessary for cracks to nucleate and the rate of propagation depend on many factors, including the corrosiveness of the solution and the stress level.

Most stress-corrosion cracking failures in reactor systems have occurred in components fabricated of austenitic stainless steel. These alloys demonstrate excellent corrosion resistance in most reactor-associated environments, but in the presence of chloride, hydroxide, or fluoride ions, stress-corrosion can occur. Because chloride ions are universal contaminants, most stress-corrosion cracks in the austenitic stainless steels are promoted by chloride ions. In the case of chloride-induced cracking the presence of oxygen is essential, and in alkaline-phosphate boiler water at 467 to 500°F, Williams and Eckel[55] have shown an approximate relationship between oxygen and chloride-ion concentration. In their study, they found that cracks developed with a chloride concentration of 10 ppm and approximately 1 ppm oxygen, but cracks were not found if the

solutions contained several hundred ppm chloride when the oxygen concentration was less than 0.1 ppm. Also, with 0.1 ppm chloride, cracks were not found when the oxygen concentration was 200 ppm. In general, the higher the temperature the greater the susceptibility of a material to cracking, but Scharfstein and Brindley[56] found cracks in cold-worked and stressed types 304 and 347 stainless steel on exposure to water containing as little as .5 ppm chloride at 165°F.

Oxygen is not necessary to produce cracks if the corrodent is free alkali;[57] however, to produce cracks at comparable temperatures, substantially greater concentrations of hydroxide ions than chloride ions are needed. With either chloride or hydroxide ions, the probability of cracking and the rate of crack propagation are greatly increased if a mechanism exists whereby the local concentration can be increased over that in the bulk of the solution, for example, by alternate wetting and drying or by the wick type of action that might occur with certain porous insulation materials.

In general, the higher the stress the greater the likelihood of cracking in a given environment, but, on the other hand, there appears to be no threshold stress below which cracking will not occur. This is illustrated by the fact that annealed and water-quenched specimens of type 347 stainless steel cracked upon exposure to a solution containing 50 ppm chloride at 250°C without the application of applied stress.[58] Apparently the residual quenching stresses were sufficient to produce cracks.

Stress-Corrosion Cracking Incidents

Reactors of nearly all types have experienced damage in the form of stress-corrosion cracking of austenitic stainless steel components. Some of the recorded incidents, and brief descriptions of the conditions under which cracking occurred are summarized in the following paragraphs.

Heat Exchangers. Instances of failures caused by stress corrosion of both types 347 or 304 stainless steel heat exchangers have been reported. In one instance a nonregenerative heat exchanger, which was part of the purification system for the primary coolant in the USS Nautilus, was shown to contain cracks upon destructive examination at Battelle Memorial Institute.[59] This heat exchanger was made entirely of type 347 stainless steel and had been in use approximately 4 1/2 years. The primary coolant entered the heat exchanger at 260°F and was cooled to 120°F. The fresh water on the secondary side entered at 110°F and left at 164°F. The secondary water normally contained about 5 ppm chloride, but occasionally it contained as much as 15 ppm; the oxygen content was 4 to 6 ppm. Although complete failure did not occur, cracks penetrated as much as 80% of the wall and undoubtedly would have lead to failure with continued use.

Stainless steel heat exchangers that remove heat by producing boiling in the secondary water are particularly vulnerable to stress-corrosion cracking in the region where the tubes are rolled into the tube sheet. If the boiling water contains even very low concentrations of chloride ions and oxygen, chloride ions can concentrate in the crevices between the tubes and the tube sheet, and cracks can be expected in the highly stressed tube ends. Tests conducted in conjunction with the Naval Reactor Program[60]

and examinations of stainless steel boilers from nonnuclear ships[61] have shown the severity of this situation.

Shortly after the startup of the Shippingport Reactor, tube failures were observed in one type 304 stainless steel heat exchanger.[62,63] Failure did not occur in the region of the tube sheet but, rather, in straight lengths of tubing. These failures were attributed to free caustic in the secondary water. Although the free caustic content of the boiler was too low to produce cracks of itself, it was postulated that steam blanketing due to an inadequate number of risers allowed concentration of the caustic to a sufficient level on the tubes to produce stress-corrosion cracks. Elimination of the free caustic from the water and installation of additional steam risers appear to have prevented any subsequent failures.

Severe stress-corrosion cracking has been experienced in at least two of the types 304 and 304L stainless steel heat exchangers used in the Savannah River Plant.[64,65] The heat exchangers operate with heavy water on the primary side and river water on the coolant side. A double tube sheet is provided at both ends to prevent leakage of moderator into the cooling water and to provide a space that permits leakage through either tube sheet to be collected. In two heat exchangers, leakage of heavy water into the space exceeded several hundred pounds per day after 4 to 4 1/2 years of service, and consequently they were removed from service. Subsequent examination showed that the tubes contained many transgranular cracks that initiated on the outer surface of the tubes. The tubes were also covered with deposits that contained substantial concentrations of chloride ions. The source of chloride ions was either decomposition of

chloride-containing gasketing material or leakage and subsequent evaporation of the river water (1 to 6 ppm chloride) through the second tube sheet. The decontamination of the damaged heat exchangers prior to repair was recently described.[66]

Fuel Elements. Although no nuclear superheat reactor has gone critical at this time, testing associated with the general nuclear superheat program has shown that, if stainless steel cladding is used for the superheater elements in conjunction with a boiling-water reactor, the probability of stress-corrosion cracking of the cladding will be high.[67] In this case the problem stems from the fact that it is virtually impossible to maintain the boiling water absolutely free of chloride ions and to prevent at least some carry-over of water with steam to the superheater elements. In addition, steam originating in a boiling-water reactor contains oxygen and hydrogen (radiolytic decomposition of water), the fuel element cladding is always stressed to some level, and, at least during startup and shutdown, a film of water is present on the cladding. Under these conditions, a high incidence of cracking would be expected. In-pile tests conducted in the SADE test loop at Vallecitos showed that deposits containing chloride ions formed on the type 304 stainless steel-clad superheater elements and, in most cases, intercrystalline cracks penetrated the cladding and allowed the escape of fission products after relatively short exposure times.[68]

Pressure Vessels. An instance of stress-corrosion cracking of austenitic stainless steel caused by fluoride ions was observed in the cladding of the S1C pressure vessel.[69] During hydrostatic testing of the vessel, a Viton-A O-ring was used as a temporary gasket. Cracks were

discovered in the vessel cladding after two weeks of exposure to ammoniated water at 480°F in the presence of oxygen. Thermal decomposition of the Viton-A released fluoride ions, and transcrystalline cracks as deep as 3/8 in. were observed in the cladding. The cracks were removed by grinding, and the metal was replaced by manual welding. The stress-corrosion cracking of austenitic stainless steel as a result of thermal decomposition of Viton-A in the complete absence of chloride ions has also been found at Bettis.[70]

Other Austenitic Stainless Components. Another failure at the Savannah River Plant occurred at a reactor outlet nozzle after about 7 years of operation.[64,65] The failure appeared as a long crack in highly sensitized type 304 stainless steel. The heat exchanger cracks referred to above were transcrystalline, but this crack was completely intercrystalline. Since the crack began on the primary water side, where the chloride concentration was less than 6 ppb, the failure was probably not attributable to chloride-induced stress-corrosion cracking. The cause of the failure has not been adequately determined. Extensive removal of the permanent biological shield and construction of temporary shielding to prevent personnel overexposure were necessary to repair the defective nozzle.

During the course of preliminary hydrostatic testing of the primary system of the N. S. Savannah, numerous leaks occurred around the austenitic stainless steel valves and pumps. At the time of testing, the primary system had been insulated but not waterproofed, and consequently the insulation, which contained some chloride, became wet. Following this the system was held at 150°F for several days and then at 510°F for

four days. Because of the fear that chloride leached from the insulation and deposited on the pipe walls might produce stress-corrosion cracking, sections of the piping were examined metallographically.[71] Although no deep cracks were found, shallow penetrations believed to be incipient cracks were observed. As a result of these findings all insulation was removed, the outer surfaces were cleaned, and indications of incipient cracks, as shown by dye-penetrant examination, were removed by grinding. Although no major material damage to the N. S. Savannah primary system occurred, failure by cracking was a distinct possibility if the chloride ions on the stainless steel surface had not been removed.

The flange leak-detector system of the second aqueous homogeneous reactor at ORNL (HRE-2), which consisted of many thousands of feet of 1/4-in.-diam type 347 stainless steel tubing, supposedly filled with pure water, developed many leaks during precritical testing.[72] The cracks were shown to be caused by chloride-ion contamination in the tubing interior. Cracking of the tubing and the high-pressure flanges was so severe that both had to be replaced, and there was a substantial delay in reactor startup.

17-4 PH Stainless Steel. Although type 17-4 PH stainless steel is not an austenitic steel, it can be hardened by suitable low-temperature heat treatment, and it possesses general corrosion resistance in aqueous environments comparable with that of the austenitic stainless steels.[73] Unfortunately, in the fully hardened condition, the material is subject to stress-corrosion cracking in high-temperature aqueous environments, even in the absence of chloride, hydroxide, or fluoride ions.[74] Overaging of the alloy results in some loss of hardness and strength but

greatly reduces the susceptibility of the alloy to stress-corrosion cracking. Stress-corrosion cracking of improperly heat-treated 17-4 PH stainless steel components in the control-rod drive system of the Dresden reactor necessitated shutdown of the reactor[75] in November 1960. Replacement of the defective type 17-4 PH stainless steel with the same alloy that had been overaged appears to have been effective in that no further difficulty has been reported.

Mitigation of Attack

There are several ways in which stress-corrosion cracking of austenitic stainless steels can be prevented. Elimination of agents responsible for cracking is one, and, at least in the case of stainless steel heat exchangers, it has been effectively applied. While it may not be possible to eliminate all traces of chloride ions from boiler water, it is possible to essentially eliminate all oxygen by the use of scavengers, such as hydrazine or sodium sulfite. In addition, free alkali is eliminated by the coordinated-phosphate water treatment in which the water is made alkaline with phosphates so that upon evaporation no free caustic exists.[76] Under these conditions cracking does not occur. In many cases it is not possible to eliminate all oxygen, and in those cases only complete elimination of chloride ions can assure freedom from cracking.

Since a tensile stress is required for stress-corrosion cracking, placing vulnerable surfaces under compressive stresses will eliminate stress-corrosion cracking. In reactor systems, however, there is little practical application for this method of control.

In common with other types of aqueous corrosion processes, stress-corrosion cracking is an electrochemical phenomenon under cathodic control. Thus, in the laboratory, cathodic protection has been shown to effectively eliminate the formation of cracks or to prevent further growth of cracks that have started. There are, however, practical difficulties in locating anodes in complicated equipment so as to ensure that adequate current reaches all surfaces that require protection.[77] Furthermore, in many systems, it is not the material that is fully submerged in the aqueous environment that needs protection, but, rather, it is the material in splash zones and similar places where normal cathodic protection cannot be extended that is the most susceptible to failure.

Perhaps the surest way to prevent stress-corrosion cracking failures in reactor environments where chloride or caustic contamination is likely is to use a material other than an austenitic stainless steel. Many of the nickel base alloys, Inconel, for example, are not susceptible to either caustic or chloride-induced failures.[78] Particularly in pressurized-water reactors, the substitution of Inconel for stainless steel in critical regions can eliminate cracking problems. (J. C. Griess)

FISSION-PRODUCT RELEASE FROM URANIUM-GRAPHITE FUELS

In a previous review article, Rosenthal[79] discussed the advantages of all-ceramic fuel elements and pointed out the severe fission-product-release problem associated with uranium-graphite fuel materials. A review of the information on the mechanisms of fission-product release from UO_2 was recently provided by Carroll.[80] Other studies, both in-pile and out-of-pile, on fuel elements for the high-temperature graphite reactor (HTGR) designed by General Atomic for the Philadelphia Electric Company and High-Temperature Reactor Development Associates,[81] the Dragon reactor,[82] the Brown-Boveri/Krupp high-temperature reactor,[83] and the pebble-bed reactor[84] have been concerned with techniques for reducing the rates of fission-product release by orders of magnitude below those observed in early screening tests. This review deals with the approaches that have been taken recently and the current status of this work.

Fuel Element Coatings

The most direct approach to the problem of fission-product release from fueled graphite is to use a gas-tight ceramic cladding on the fuel element. Unfortunately every element with an impervious coating tested to date has failed under irradiation. One of the more promising techniques involved coating the element with siliconized-silicon carbide. This coating was demonstrated to be protective to graphite for 1000 hr in air at 1800°F. Under irradiation, however, the coatings have failed consistently from the combined effects of thermal cycling and irradiation. Fuel elements with low-permeability pyrolytic-carbon coatings have suffered the same fate. Apparently the cracks that form in the

graphite under the coating as a result of thermal stresses or irradiation-induced contraction readily propagate through bonded coatings. Once the coatings fail the fission-product release rates are the same as those of uncoated fuel elements under the same conditions. This feature is one of the major drawbacks to the concept of fuel-element coatings. There is still hope that an acceptable coating will be developed, but no reactor materials development group is now placing major emphasis on this approach.

Low-Permeability Bottles

A second way to control fission-product release is the use of a low-permeability graphite bottle. This approach has been selected for the reference design for the Brown-Boveri/Krupp reactor, and low-permeability graphite sleeves are used for the HTGR and the Dragon reactor. Ordinary reactor-grade graphite has a permeability of about 1 cm^2/sec. Fission gases flow readily through this material by combined viscous and Knudsen flow. The "dusty gas" theory of Evans *et al.*[85] is quite successful in describing the combined processes quantitatively. By a process of repeated impregnations of this type of base stock with organic liquids, such as furfuryl alcohol, followed by thermal decomposition treatments, the permeability can be reduced to values of the order of 10^{-8} to 10^{-10} cm^2/sec. In the low-permeability material the migration of fission gases occurs primarily by Knudsen flow and is sufficiently slow that most of the short-lived volatile precursors decay to long-lived stable daughters before they migrate out of the fuel element.

A typical chain in which decay is important is

$$16\ \text{s}\ Xe^{140} \rightarrow 66\ \text{s}\ Cs^{140} \rightarrow 12.8\ \text{d}\ Ba^{140} \rightarrow$$
$$40.2\ \text{h}\ La^{140} \rightarrow \text{stable}\ Ce^{140}\ .$$

Both Ba^{140} and La^{140} are energetic gamma emitters, and Ba^{140} is long lived. In ordinary graphite, a volatile precursor of the undesirable Ba^{140} may escape and subsequently deposit as Ba^{140} on the heat exchanger after decay of the precursor. If, however, the volatile Xe^{140} and Cs^{140} decay to Ba^{140} before escape from the low-permeability bottle, then the activity will be retained in the fuel element, unless the temperature is sufficiently high that the barium will vaporize. Other decay chains with volatile precursors and long-lived daughters are those of mass 90, 91, and 137. Even though the short-lived species decay before they reach the external surface of the fuel, stable xenon and krypton can slowly escape. In this way the internal pressure will not become sufficiently great to cause the bottle to rupture, unless a severe upward temperature transient occurs. A quantitative treatment of the relationships between permeability and release rates of species with different half-lives has been given by Prados.[86]

There are a number of disadvantages to the low-permeability bottle approach. One is that the fuel elements are unprotected if the bottle cracks. A second is that the low-permeability bottle must be sealed after the fueled body is placed inside it. Seals have been made in two ways: (1) by brazing the joint between low-permeability components with a brazing alloy such as that developed by Donnelly and Slaughter[87] and (2) by repeatedly reimpregnating the graphite after the fueled core has

been placed within a machined graphite ball and the end plug screwed in. The second technique is to be used for the Brown-Boveri/Krupp reactor. Fuel elements with brazed end plugs have invariably leaked under irradiation. Elements made by reimpregnation have not been adequately tested. Based on the meager information available, it is concluded that further work will be required to properly assess the usefulness of low-permeability bottles.

Coated Particle Studies

A third technique for retaining fission products in an all-ceramic fuel element involves the use of coated fuel particles. In this case individual fuel particles, such as UO_2 or UC_2, are coated with an impervious coating before they are incorporated into a fuel element. In this way each fuel particle is protected individually and the failure of any one coating results in a minuscule release of activity from the fuel element. This approach has the additional advantage that a fuel which is unstable in air, such as $(U,Th)C_2$, is protected from the atmosphere during the fabrication of the fuel body and during subsequent storage. For some fuel elements, coatings on the carbide particles are desirable for the latter reason, even if the coatings are unable to survive the irradiation environment.

Although many types of coatings have been tried, the chief emphasis to date has been on two types: Al_2O_3 for UO_2 and pyrolytic carbon for UC_2. Like other oxide coatings, Al_2O_3 is applied to UO_2 by the hydrolysis of $AlCl_3$ at high temperature.[88] Pyrolytic carbon is applied by the pyrolytic decomposition of organic gases, such as methane or acetylene.[89]

Depending on the temperature and rate of deposition of the pyrolytic carbon, coatings may be made with different characteristics. The two most common types of coatings are referred to as laminar, or "onion-ring," and columnar.[90] The laminar coating is characterized by its fine-grained, smooth, layer-like structure, whereas the columnar coatings are distinguished by large, radially oriented grains and rough surfaces. Some coated-particle vendors have chosen to apply an inner laminar coating followed by a columnar coating. The composite structure is referred to as "duplex" because of the different appearance of the two layers.

The impetus toward the coated-particle approach was provided by the SP-5 sweep capsule test at Battelle Memorial Institute for the Sanderson and Porter pebble-bed reactor program.[91,92] The fuel element was a 1.5-in.-diam ball of graphite that contained 100- to 140-μ-diam UO_2 particles coated with a 40-μ thickness of vapor-deposited Al_2O_3. During irradiation, a helium sweep was monitored periodically to determine the extent of fission-product release. At a fuel temperature of about 1350°F, the initial fission-gas release rate was about 10^{-7} of the amount produced. Cracking of the particles was first noted at 1 at. % burnup and became relatively severe at between 3 and 5 at. % burnup. After about 6 at. % burnup, the postirradiation examination showed that 8% of the particle coatings had cracked. The final fission-gas-release rates were about 10^{-2} of the amount generated for the long-lived gases and below 10^{-4} for most of the shorter lived ones. The fact that 92% of the particle coatings did not fail spurred efforts toward improved particle coatings.

One of the questions that arose during irradiation of the SP-5 Capsule was the compatibility of Al_2O_3 with graphite. Above about 2000°F

these materials react to form CO and other reaction products that are volatile. In order to achieve an inherently compatible system, attention was directed toward the thermodynamically stable system, pyrolytic-carbon-coated uranium carbide.

Screening tests at General Atomic,[93] Battelle Memorial Institute,[94] and the Oak Ridge National Laboratory[95] showed that uranium carbide particles with pyrolytic-carbon coatings maintained their integrity when tested by acid leaching, alpha contamination, and neutron activation. High-quality particles released 1 to 10 ppm Xe^{133} in 24 hr at 1832°F. Even at 3300°F, for periods up to 100 hr, the Xe^{133} release was found to be small.[95] Above about 3600°F, particles from some batches were observed to rupture spontaneously. This rupturing was attributed to stresses on the coatings which could result from differential thermal expansion between the uranium carbide core and the pyrolytic-carbon coating.

Although the pyrolytic-carbon coatings retained xenon at 3300°F and below, Ba^{140} and Sr^{84} were released rapidly at 2550°F and above;[95-97] I^{131} and Te^{132}, like Xe^{133}, were released only when defective coatings were present. A number of studies are under way at several installations to determine the rates and mechanisms of migration of barium, cesium, and strontium through pyrolytic graphite.

The observations of the migration of barium led to the concern that uranium might also migrate through the coatings. To check this possibility, unsupported coated particles were heat treated and then examined by neutron-activation tests and by low-voltage radiography. The tests showed that uranium did migrate through both laminar and columnar coatings at 2900°F.[97] In neither coating was the migration uniform, and

concentration of the uranium at certain points suggested that the migration occurred by diffusion along defects. In laminar coatings uranium was observed to spread circumferentially along laminations, resulting in moon-shaped regions of high uranium concentration. In samples of both types of coated particles, a few coatings were found that were free of uranium. This observation leads to the hope that a suitable fabrication technique can be devised to prevent the migration. It is also possible that uranium migration will not be a problem at lower temperatures, for example, 1800°F.

Irradiation Tests on Coated Particles

Irradiation tests on unsupported coated particles and fuel elements containing coated particles have been carried out at Battelle Memorial Institute, General Atomic, and the Oak Ridge National Laboratory. Tests have also been made on uncoated particles for comparison. Both sealed-capsule and sweep-capsule experiments have been run. In the latter type of experiment, the rates of release of the noble gases with half-lives longer than about 1 hr are monitored periodically. By means of "daughter traps" developed at Battelle Memorial Institute[91] information has also been obtained on the rates of release of very short-lived noble gases, such as 16 s Xe^{140}. After irradiation the fuel samples are examined visually and metallographically for evidence of cracked coatings, fuel migration, and changes in fuel microstructure.

Sweep capsules are especially effective for assessing the fission-gas-release characteristics of fuel materials during irradiation, since the effects of temperature, thermal cycling, and burnup can be determined.

Data are normally reported in terms of the R/B ratio, which is the rate of release of a given fission product divided by the rate of formation of that species. During startup and after transients the theoretical value of the R/B ratio is difficult to compute. At steady state, however, that is, after irradiation for about five half-lives of the particular species being monitored, the R/B ratio for a given species should be constant. For the two known processes of fission-gas release in-pile, that is, recoil and diffusion, the R/B ratios for uniform spheres of radius, a, are given by the following equations.

For recoil,

$$\frac{R}{B} = \frac{3}{4}\frac{d}{a} - \frac{1}{16}\left(\frac{d}{a}\right)^3 , \tag{1}$$

where d is the recoil range in the fuel.

For diffusion,

$$\frac{R}{B} = 3\left(\frac{D}{a^2\lambda}\right)^{1/2}\left[\coth\left(\frac{\lambda a^2}{D}\right)^{1/2} - \left(\frac{D}{a^2\lambda}\right)^{1/2}\right] \qquad \text{for } \frac{\lambda a^2}{D} > 100 . \tag{2}$$

In Eq. (2), D is the diffusion coefficient and λ is the decay constant. For small values of R/B, Eq. (2) may be approximated by

$$\frac{R}{B} = 3\left(\frac{D}{a^2\lambda}\right)^{1/2} . \tag{3}$$

It is useful to compare ratios of R/B for different isotopes of xenon and krypton being released simultaneously. For the recoil process, values of R/B should be independent of λ. When diffusion occurs, R/B should vary inversely with the square root of λ. Thus a plot of log R/B vs log λ should yield a straight line with a slope of -1/2. This latter condition

is normally observed in data from tests on both coated and uncoated uranium carbide particles, suggesting that the mode of release is diffusion. The diffusion is not necessarily out of the fuel particle. Some evidence suggests that the rate-controlling process at temperatures up to 1832°F is the diffusion of recoils out of the graphite surrounding uncoated particles or particles with defective coatings.

Typical release-rate data obtained by Carroll and Reagan[98] for uncoated and coated particles at 1500°F are shown in Table II-1, together with data on single-crystal UO_2 for comparison. The particles were also tested at other temperatures, and the temperature dependence of release was found to vary from sample to sample. The sample with a laminar coating showed almost no temperature dependence, but the sample with a columnar coating indicated a marked temperature dependence. Apparent diffusion coefficients of uncoated particles of UC_2 showed an activation energy of about 20 kcal/mole and those of the UO_2 showed an activation energy of 30 kcal/mole. In both UO_2 and UC_2 samples, I^{133} accounted for about 25% of the Xe^{133}, and I^{135} accounted for about 80% of the Xe^{135}. Since the activation energy for iodine release was about the same as for xenon release, the mechanisms of iodine and xenon release appear to be identical.

The release-rate characteristics of the sample that had a duplex coating were particularly encouraging. After a burnup of 14.7 at. % of the total uranium, the release rate was about 10^3 less than that of uncoated particles under the same conditions. Postirradiation examination showed extensive damage to the inner, laminar coating, but the outer coating was undamaged. If the cause for the presence of 0.3% defective coatings in this material can be established, the release rates may be reduced even further.

Table II-1. FISSION-GAS-RELEASE DATA FOR COATED AND UNCOATED FUELS TESTED AT 1500°F

Fuel	Type of Coating	Fuel Particle Size	Ratio of Release Rate to Generation Rate, R/B					Defective Coatings[a] (%)
			Kr^{87}	Kr^{88}	Kr^{85m}	Xe^{135}	Xe^{133}	
			$\times 10^{-3}$	$\times 10^{-3}$	$\times 10^{-3}$	$\times 10^{-3}$	$\times 10^{-3}$	
UO_2[b]	None	50-mil plate	0.81	1.3	1.5[c]	1.2	3.4	
UC_2	None	175–246 μ	3.8	5.3	10[c]	10	72	
UC_2	Laminar	175–200 μ	0.18	0.18	0.27[c]	0.20	0.50	5
UC_2	Columnar	150–250 μ	0.024	0.027	0.057[c]	0.058	0.33	2
UC_2	Duplex	175–246 μ	0.003	0.004	0.06[c]	0.044	0.034	0.3

[a]Based on postirradiation visual inspection.

[b]Single-crystal sample.

[c]Based on a Kr^{85m} yield of 1.5%.

Although the results of the sweep-capsule tests at 1500°F were encouraging, limited tests on fuel elements at higher temperatures have not been so favorable. Evidence for this is shown in Table II-2. The comparisons are based on Kr^{88} because it has no troublesome precursors and its specific activity is high. The minimum values in Table II-2 occurred during or shortly after startup. At temperatures of 2000 and 2600°F, the rates of release increased approximately exponentially with burnup. Duplicate unsupported particles in sealed capsules showed much lower incidences of failure. In the sweep capsules, tantalum sheaths on thermocouples were found to be severely oxidized, indicating that the sweep gas contained oxidizing impurities. Thus the failures observed may not have been due to irradiation effects alone. In capsule ORR-08A-5, the fuel elements containing particles with duplex coatings were affected by irradiation at 1750°F far more than the unsupported particles with duplex coatings. It is important to note that because of differences in deposition conditions, the two batches of particles were not identical. For the unsupported particles, for example, the nominal thickness of the inner laminar layer, which is exposed to fission recoils, was 25 μ. For those in the fuel elements, however, this thickness was only 12 μ. In addition, more temperature changes and thermal cycles were experienced by the graphite-matrix fuel elements than by the unsupported particles.

Recent irradiation tests performed at Battelle Memorial Institute on Al_2O_3-coated UO_2 have identified a major problem with this system.[99] Samples that were irradiated to burnups of 2 to 4% of the uranium atoms at 212°F showed a much higher incidence of coating failures than duplicate samples irradiated at 1960°F. A possible cause for this effect is

Table II-2. SUMMARY OF RESULTS FROM FISSION-GAS-RELEASE EXPERIMENTS AT ORNL ON FUEL ELEMENTS CONTAINING COATED URANIUM CARBIDE PARTICLES

Experiment Designation	Type of Coating	Temperature (°F)	Burnup (%)	Ratio of Release Rate to Birth Rate for Kr^{88}, R/B	
				Minimum	Maximum
MTR-48-5	Laminar	2600	7	5×10^{-4}	2×10^{-2}
MTR-48-6	Laminar	2000	1.7	1×10^{-4}	40×10^{-4}
ORR-08A-5	Duplex	1750	9	1×10^{-5}	1×10^{-3}

the observation of Bleiberg and co-workers that Al_2O_3 loses its crystallinity on exposure to fission fragments at low temperatures.[100] At higher temperatures the damage is annealed out without loss of crystallinity. If loss of crystallinity is the cause of low-temperature failures of Al_2O_3 coatings, failure might be prevented by the use of an undercoating of a nonfissile ceramic to catch the fission fragments.

Fission-Product Control by Purging

In the HTGR and the Dragon reactor fuel elements, a low-permeability graphite sleeve is used to keep fission products out of the coolant. Between the sleeve and the fueled core is an annulus through which helium is passed continuously. After leaving the fueled core, the helium sweep gas passes through a silver-coated charcoal trap at the bottom of the fuel element and then into a fission-product clean-up system outside the reactor. Coated particles are used in the fueled core of the HTGR, not to retain fission products, but, rather, to prevent hydrolysis of the $(Th,U)C_2$ during fabrication and handling.

Irradiation tests on the HTGR fuel element were conducted in the GAIL loop in the General Electric Test Reactor (GETR) at Vallecitos.[101] All conditions were those of the reference design, except that the fuel element length was one fourth that of the reference design. The graphite sleeve had a room-temperature permeability of 10^{-5} cm^2/sec. Both the primary coolant stream and the purge line were monitored for radioactive noble gases. The results are still tentative and have not been published, but it is known that the observed activity levels were fully acceptable for the HTGR.

One other problem has arisen, however, that reflects on the acceptability of the purge technique for controlling fission-product release. Recent work of Walton[102] has shown that cesium, strontium, and barium migrate rapidly through low-permeability, impregnated graphite at rates that are independent of the permeability with activation energies of only 7 to 18 kcal/mole. The rates are consistent with an in-pore surface diffusion model and are several orders of magnitude greater than the rates at which these species escape from lattice positions into which they have recoiled. Thus while the purge is effective in reducing contamination from daughters of volatile nuclides, a mechanism exists for contamination of the primary coolant. The severity of the problem depends on the fugacities of the fission-product species exposed to the coolant. Further work will be required to assess the magnitude of the problem. (J. L. Scott)

POWER REACTOR STABILITY EXPERIENCE

The problem of power reactor stability continues to receive considerable attention, with progress ranging from refinements in theoretical treatments to improvements in experimental techniques. Measurements of system stability characteristics are now regarded as being as much a part of the essential reactor startup experiments as the usual determinations of critical loading, available excess reactivity, rod calibrations, and flux distributions. Techniques have been developed that provide detailed stability information from simple experiments; and, by noise analysis, basic information on reactor properties can be obtained with no interference with reactor operation.[103]

Fuel-Rod Bowing in Sodium Reactor Experiment

Despite the refinements in methods of prediction and the inclusion of stability measurements in startup programs, there are occasionally surprises when a reactor is taken to power, inasmuch as effects that were either unforeseen or unappreciated become important. Operation of the SRE with its second core is an illustration.[104]

When the power was raised for the first time after the second core was installed, it was found that abrupt power changes resulted from small rod motions and that smooth control was impossible. The first core had shown excellent self-regulation at power, and on the basis of predictions and low-power tests the second core was expected to be stable. An experimental program was therefore initiated for finding the cause of the difficulty.

The first step was the use of an on-line analog computer that simulated the reactor system with two feedback loops. The reactor was operated in a range of moderate power while the rod positions were transmitted to the simulator. The output of the simulator and the reactor nuclear power were recorded for comparison, and the time constants and gains of the feedback loops were adjusted until the simulator output matched the behavior of the actual power. A positive feedback was found to be necessary, indicating that despite a predicted strong, negative Doppler coefficient, there was a fast, positive power coefficient. Doubts as to the uniqueness of the simulator results were resolved by experiments with power ramps, flow ramps, and ramps in both power and flow, which showed both the expected Doppler coefficient and a predominant effect, which was positive.

These results led to calculations which indicated that the probable cause was outward bowing of the rods in a fuel cluster when the power was raised.[105] No similar effect was observed with the first core, mainly because smaller clearances permitted less bowing of the fuel rods. Rod bowing was confirmed when tests showed that the net fast power coefficient decreased and became negative as fuel clusters were progressively replaced with clusters restrained from bowing by wire wrapping.

In the latter phase of the experiments, power ramp tests were analyzed by feeding the reactor chart records into the analog computer and varying the number and characteristics of the feedback loops until the simulator output matched the reactor output. The results of these tests proved valuable in separating various effects and required little reactor operating time.[106]

Stability of Direct-Cycle Turbojet-Reactor Systems

Reactors were joined to turbojet engines in the HTRE (Heat Transfer Reactor Experiment) series. Air entering through an axial-flow compressor passed through the core to a turbine, which was directly coupled to the compressor, and out through a tailpipe and nozzle.

This unusual reactor application was susceptible to an unusual form of instability.[107] In the core, the specific heat input to the air depended on the air flow rate; in the engine, the turbocompressor speed and air flow depended on the specific heat input to the air. Stability analyses and operating experience showed that HTRE-3 was stable during high-speed, high-power operation; but when the speed was reduced, a point was reached where the engine-reactor system became unstable, and the speed dropped rapidly.

The low-speed instability was normally circumvented by the control system, which used combustion heat to control the air temperature entering the turbine while holding the nuclear power at a setpoint. But on three occasions, the speed was brought up through the region of instability on nuclear power alone, with manual control. This was a delicate operation, requiring close control of speed and fuel-plate temperatures within the proper relationships, but it was judged to be feasible as a routine method for starting a reactor-turbojet system.[108]

Transient Analyses

The experimental program of the Yankee reactor plant included tests that confirmed the safe operation of the plant for accident transients. In addition, the stability of the plant for normally controlled loading

and unloading was demonstrated. Comparison of predicted and actual response was difficult because, for conservatism, the most adverse combinations of reactivity coefficients had been used in the analytical studies. Also, feedwater transients were not included in the analysis, and while these proved to increase the stability for normal load changes, in cases of turbine trip or reactor scram, the transients in feedwater flow caused more severe primary-loop temperature transients than predicted. Additional feedwater control was added as a result of this experience.[109]

A report by the operators of the Shippingport pressurized-water reactor describes in detail the analysis of transients from chart records obtained during two unscheduled safety shutdowns of the plant.[110] During the first shutdown, which was caused by a faulty valve-position indicator, all variables remained under control. The second shutdown occurred during a test in which the turbine power was being rapidly reduced. Burnup had resulted in withdrawal of control rods to a region of low sensitivity, and the temperature coefficient of reactivity was reduced for the same reason. As a result, although the rods were being driven inward, the coolant temperature rose to the trip point and caused a rod drop.

Analyses of the chart records for these unscheduled transients were hindered by the difficulty of reading and correlating time on the chart records from various recorders in the plant. This is a common experience of those who seek to extract information on fast-moving incidents from slow-moving charts designed for another function in routine plant operation.

Noise Analysis

The small fluctuations in the nuclear power of a reactor, when properly analyzed, can afford information on the inherent stability of the system. This is so because the frequency spectrum of the fluctuations (the reactor-spectrum driving function) shows the response of the system to the small input disturbances, or noise, which are always present. For a thorough treatment of random processes in control theory, see Laning and Batting.[111]

Determinations of the reactor transfer function by using noise analysis to obtain the power spectral density have been based on the assumption that the reactor-spectrum driving function is a white noise, that is, a random disturbance possessing all frequencies in equal amounts. The SRE has been used in a program of improvement of techniques for the measurement of power spectral density. Reactor flux spectra were measured at three power levels, and the frequency response was compared with the response to reactivity oscillation under identical operating conditions. The results indicated that the reactor-spectrum driving function deviated from white noise at some low frequency and that the frequency where deviation occurred was higher at elevated power and temperature.[112]

The power spectral density of the N.S. Savannah reactor has also been measured, in a search for plant instabilities, with the ship at dockside, in a calm sea, and in a heavy sea.[113] A significantly higher amplitude of the power spectral density, around 1 cps, was observed when the sea was rough. Ship heave, roll, and pitch would be expected to affect inertial and thermal driving heads of the water in the core.[114]

Xenon Instability

The Shippingport reactor, during operation with the first set of enriched seed elements, showed a tendency to develop oscillatory power distributions because of xenon instability. The second and third seeds differed from the first in increased fuel loading and the use of two fuel zones, with boron poison in one zone. No spontaneous radial power oscillations occurred during seed 2 and 3 operation, and deliberate attempts to induce oscillations by asymmetric rod motion resulted only in highly damped oscillatory behavior. Fluctuations in controlling-rod-group height, possibly caused by axial xenon oscillation, were observed, however, at certain times during seed 2 full-power operation.[115]
(P. N. Haubenreich)

LOSS OF COOLANT ACCIDENTS IN GAS-COOLED REACTORS

(Held over from Vol. 4, No. 2)

IN-PILE LOOP HAZARDS

(Held over from Vol. 4, No. 2)

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III. CONTROL AND INSTRUMENTATION

POSTINCIDENT ANALYSIS OF SOME ASPECTS OF THE SL-1 CONTROL-SYSTEM DESIGN

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(Editor's Note: The role of the control system in the SL-1 incident has been of considerable concern to designers of reactor systems. The following review was prepared by the organization responsible for the design and initial operation of the SL-1 and, although quite candid, is based upon a control philosophy which is less restrictive than that advocated by the editors. Thus the ideas and conclusions expressed herein do not necessarily represent the views of the editors of Nuclear Safety nor do they necessarily reflect the official position of the AEC, except where the Board of Investigation or Commission officials are quoted or referenced.)

Sufficient evidence[1-3] is now available to state, with a reasonable degree of certainty, that the incident which destroyed the SL-1 on

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January 3, 1961, was a nuclear excursion caused by manual withdrawal of a single control rod (the central rod) to an indicated position of approximately 20 in. (See article in Section VI of this issue on the Final Report of the SL-1 Board of Investigation.) At the time of the incident, the over-all control system for the reference 3-Mw (thermal) core of 40 fuel assemblies comprised five movable cross-shaped control rods, six fixed strips of cadmium inserted into two tee-shaped control-rod channels, full-length and half-length burnable-poison strips attached to the fuel assemblies, and a manually operated soluble-poison backup shutdown system.

The purpose of this analysis is to re-examine the safety aspects of the control systems employed in the light of lessons learned from the SL-1 incident and discussions stimulated by it. The lessons learned can be applied to the design, installation, operation, and maintenance of more general movable-control systems, control rods, or other control devices for reactors.

Reactor Components

Fuel Assembly. The reference core included 40 fuel assemblies, each containing nine fuel plates. The active region of each plate consisted of a uranium-aluminum-nickel alloy core (0.050 in. thick, 3.50 in. wide, and 25.8 in. long) clad with a 0.035-in. thickness of aluminum-nickel alloy (X8001). The uranium was enriched to approximately 93 wt % U^{235}, and the nickel constituted approximately 1.6 wt % of the fuel-plate core (see ref. 2, Fig. 24).

Burnable-Poison Strips. To assist in control of the excess reactivity in the core, thin burnable-poison strips were fusion welded to the side plates of each fuel assembly. The strips were extruded from a mixture of aluminum, nickel, and boron highly enriched (>90%) in B^{10}. One full-length (25.8-in.) poison strip, containing approximately 0.5 g of B^{10}, was welded to a side plate of each fuel assembly. One half-length (12.9-in.) strip, containing approximately 0.2 g of B^{10}, was welded to the bottom half of the other side plate of each of the 16 central fuel assemblies. Burnable-poison strips of a similar type had been used successfully in BORAX III (see ref. 2, p. 216).

Core Shroud. The core shroud structure, fabricated from aluminum-nickel alloy (X8001), provided for a maximum loading of 59 fuel assemblies and one neutron source assembly into 16 compartments. Twelve of the compartments (~8 in. square) were designed to accommodate four fuel assemblies, and the four corner compartments were capable of containing three assemblies (see ref. 2, Fig. 22). In the reference core of 40 fuel assemblies, dummy fuel assemblies were added to maintain vertical alignment of the fuel assemblies and the source assembly in their respective compartments.

The walls (0.156 in. thick) of the compartments formed channels (0.5 in. thick) to guide the vertical movement of five reference cross-shaped control rods and four additional tee-shaped rods. An average water clearance of 0.14 in. existed between the guide shrouds and either side of a control rod.

Control Rods. Each of the five cross-shaped control rods consisted of a cadmium absorber section (0.060 in. thick, 14 in. wide, and 34 in.

long) clad with 0.080-in.-thick aluminum-nickel alloy (X8001). The total rod thickness was 0.220 in. Although the active core length was only 25.8 in., the additional absorber length (more than 8 in.) of the control rods was provided to give essentially full coverage of the core during normal shutdown and during assembly (or disassembly) of a drive to its control rod (see Fig. III-1). The central rod had a 17-in.-long follower. The other four rods had 5-in.-long followers. The upper end of each rod had a 3-in.-span cross extension and a stainless steel ball-joint end fitting. The end fitting provided a simple means of connecting the rod to the drive mechanism (see ref. 2, Fig. 27).

Control Rod Drives. The control rods were driven independently by rack-and-pinion-type drive mechanisms (see Fig. III-2). The rack and pinion, pinion-support bearings, and rack backup roller operated in the saturated-steam and water atmosphere above the reactor vessel. The rack-and-pinion teeth were chromium plated. The pinion-support ball bearings were of Stellite. Approximately one year before the incident, they were replaced with standard-alloy carbon-steel ball bearings (see ref. 3, p. 67).

Soluble Poison System. There appeared to be no reactivity problem in shutting down the reactor with the system of control rods (even if one control rod were fully withdrawn) if the reactor was maintained at design temperature and pressure. In order to maintain shutdown at ambient temperature, however, at least partial insertion of the central rod was always necessary.

A backup shutdown system, using a relatively concentrated solution of boric acid (100 g of H_3BO_3 per gallon), was available to provide

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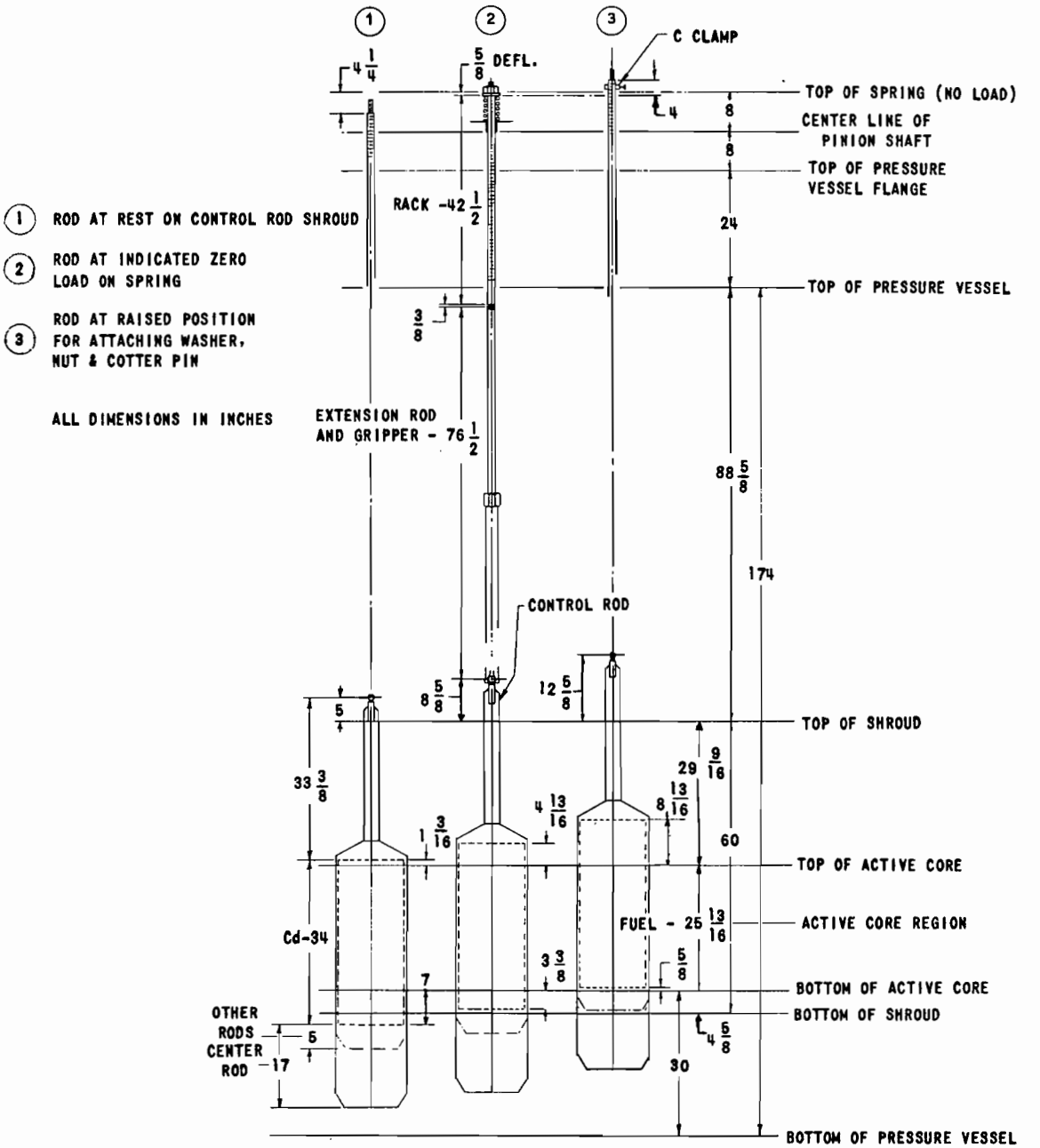

Fig. III-1. SL-1 Control Rod Positions.

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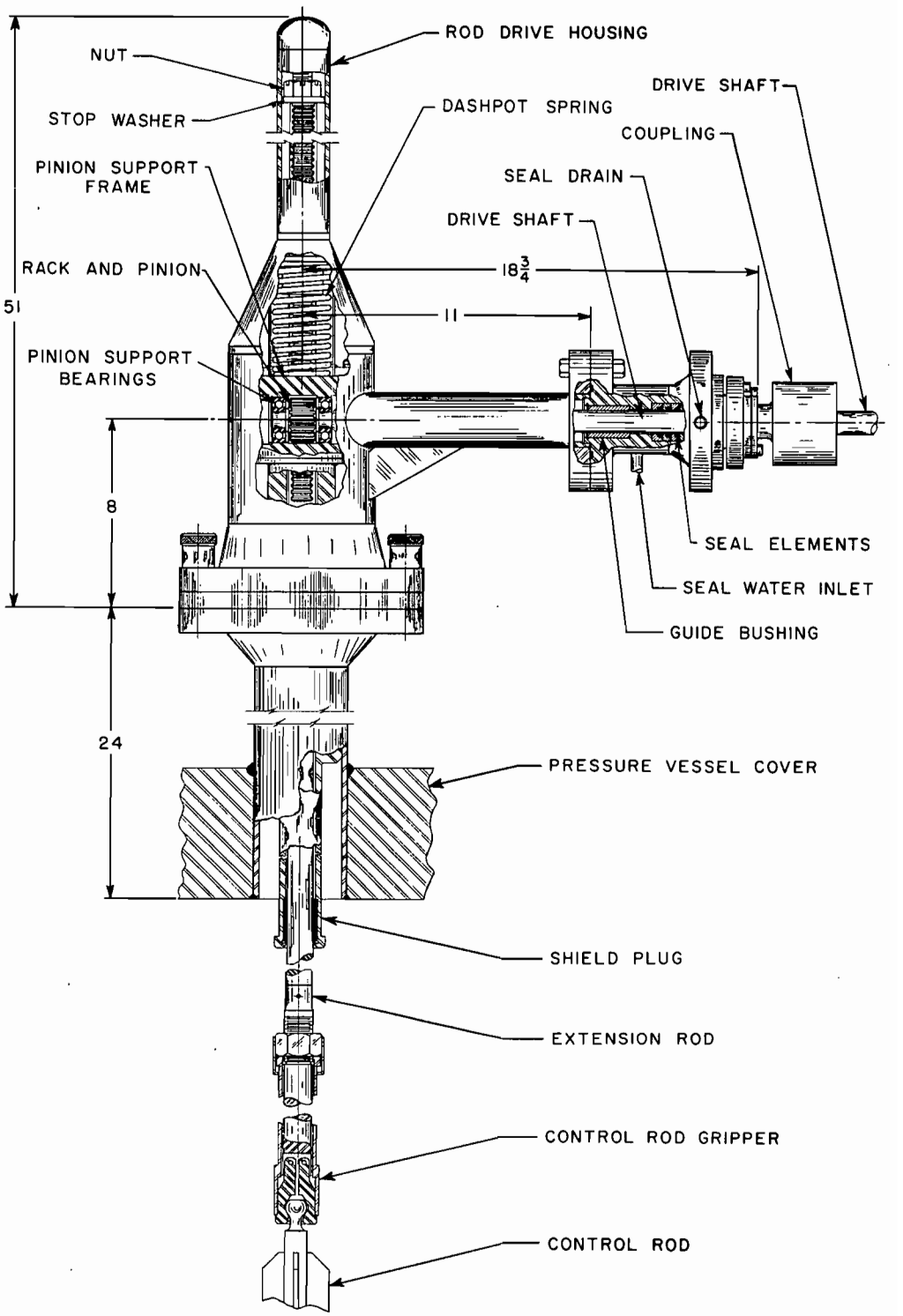

Fig. III-2. SL-1 Control Rod Drives.

additional control, if needed, as the temperature of the reactor water decreased. The system included a hand-operated pump to be used when the reactor was at above-atmospheric pressure. At atmospheric pressure, the solution could flow into the vessel by gravity.

The boric acid solution was available as an auxiliary shutdown method and was to be used at the discretion of the operator. The system was not intended to effect an emergency shutdown. Rather, it was to be available for such contingencies as the inability to insert a control rod. In the SL-1, the boric acid system was not employed for reactivity control except for a period of experimentation and for a short xenon-measurement run at power.

Preincident System Malfunctions

Deterioration of Poison Strips. In August 1959, some bowing of the burnable-poison strips was observed between the fusion welds that held the strips to the side plates of fuel assemblies. After 400 Mwd of reactor operation (February 1960), it became evident that the actual rod bank position, at power, was dropping much more rapidly than was predicted by the calculations (see ref. 3, Fig. 30). It was believed that the progressive lowering of the critical rod bank position during reactor operation resulted from a more rapid loss of B^{10} from the reactor core than had been anticipated.

During a periodic inspection of fuel assemblies in August 1960, it was discovered that the burnable-poison strips were deteriorating. Removal of fuel assemblies from the center of the core was accomplished with considerable difficulty. These fuel assemblies were bound tightly

within their compartments, probably because of the extreme bowing and deterioration of the poison strips. Remote viewing of strips indicated that the bowing was more pronounced in the center of the reactor core, where the power density was higher (see ref. 3, pp. 31–32). To compensate, in part, for this stationary-control loss, three cadmium blades were added in November 1960 in each of two diametrically opposite tee-rod locations. These served as fixed control rods in the reactor core (see ref. 3, p. 52).

Early in the postincident investigation, it was suggested that the bowing of the poison strips might have led to bending of the control-rod guide shrouds and a resultant narrowing of the control-rod channels. This was previously listed as a possible cause of sticking or binding of control rods, but it is no longer offered as an explanation of the "rod sticking" experienced during reactor operation. Although the deterioration of the burnable-poison strips was not anticipated and was quite undersirable, it appears to have been irrelevant to the actual SL-1 incident.

Sticking of Control Rods. During reactor operation, a history of 84 occasions of "sticking of the control rods" was accumulated. This "sticking" has been categorized into three types (see ref. 3, p. 61):

> "Type I – Sticking of a control rod resulting in failure to meet the drop time requirements (one second for 10" drop; two seconds for a 30" drop) and which did not require a power assist from the drive assembly.
>
> "Type II – Sticking of a control rod in which the control rod stopped and required a power assist to enable the control rod to reach its zero position (even if it subsequently fell free at a lower level).
>
> "Type III – Sticking of a control rod in which it was not possible to drive the control rod in a desired direction, e.g.,

> clutch slippage during a rod withdrawal, or failure of a drive assembly shear key or gears resulting in failure to drive a control rod."

There were 19 Type-I, 52 Type-II, and 13 Type-III sticking events.

There is no record of measurements having been made, with the reactor at operating temperature and pressure, to check the dimensions of the pressure-vessel head with regard to alignment of the drive mechanisms and the control rods. Misalignment of a control-rod drive relative to the control-rod guide shroud is still a reasonable possibility; however, it follows that sticking should then have occurred in a much greater percentage of the rod movements.

As mentioned earlier, bowing of the burnable-poison strips appears not to have caused "rod sticking." In November 1960, a "burr" was observed and was removed from one control-rod guide shroud (see ref. 3, p. 63). This did not eliminate subsequent sticking in this location. Indeed, out of a total of 20 sticking events during the period May 1959 to December 1960, nine sticking events occurred at this location within one and one-half months after removal of the burr (see ref. 3, Appendix A).

Postincident examination of the shrouds and the control-rod blades has revealed that only a thin film of corrosion was present. Rub marks and scratches found on the guide shrouds and control-rod blades are attributed to normal wear during preincident operation. There was no evidence of inward bowing or warping of the shroud, significant loss of cadmium, abnormal wear of rods or shrouds, or buildup of crud in the control-rod channels. In short, the shrouds and control rods appear to have been in normal condition prior to the incident, after more than two years of use.[4]

Among other possible causes of what has been termed "control-rod sticking" are (1) crud in the rod-drive seal, (2) crud in and wear of the Stellite pinion-support ball bearings, (3) crud in and wear or corrosion of the replacement carbon-steel ball bearings, and (4) a "hydraulic-piston" effect, that is, a momentary partial vacuum in the space above the rack as the rod was dropped.

The degree of the "hydraulic-piston" effect is influenced by such parameters as reactor pressure and water temperature, seal-water inlet temperature, and water level in the rack housing. This effect had been noticed during the testing of a prototype SL-1 rod-drive mechanism at the Argonne National Laboratory. (A piston effect of this type has since been observed on a similar drive mechanism designed for another reactor.[5]) Consequently, the washer on top of the rack was modified (reduced slightly in diameter) to permit more rapid equalization of pressures between the reactor and the top of the rack housing. This modification appeared to have eliminated the effect, as evidenced by further testing.

In summary, no one cause of the sticking events has been established. More likely, there were several contributing factors.

Assembly of Drive to Control Rod

Design Features. As designed, the cold, fresh, xenon-free SL-1 reactor could achieve criticality with the central control rod raised to an indicated position of about 19 in. and with the other four cross-shaped control rods fully inserted (at indicated zero position). This fact was recognized by the designers and the operators. Several design features were incorporated to ensure the safety of personnel during

assembly (or disassembly) of the drive mechanisms to their respective control rods. The design of the absorber section of the rods included extra cadmium so that slight raising of the rods resulted in a negligible increase in reactivity. Moreover, the likelihood of a fast rod withdrawal into the region of supercriticality was decreased by the weight of the mass being lifted (85 lb in water) and the array of equipment installed on the pressure vessel head, which tended to restrict bodily movements.

Procedure. The normal procedure for assembly or disassembly operations was to work with only one control rod at any given time. The other rods were to be fully inserted. The normal procedure required the cooperative efforts of two men; one to lift or hold the rod and the other to attach or to remove a C-clamp. During disassembly, after removal of the seal and rack housing, the control rod was supported by the internal dashpot spring. The next step was to raise the rod a short distance in order to attach a C-clamp to the rack to prevent the rack from dropping while the nut and washer were being removed from the top of the rack (see Figs. III-1 and III-2). Subsequently, the C-clamp was to be removed and the control rod lowered to its rest position. The order of assembly of the drive to its rod was essentially the reverse of the order of disassembly.

The instructions took into account the hazard of raising the central rod too far. It was stated explicitly that the rod was to be raised "not more than 4 inches" (see ref. 3, p. 57). Very little reactivity would have been added, however, if all five cross-shaped control rods had been at the indicated 4-in. position, and the control margin was sufficient to override this small extra reactivity. At this point,

because the absorber overlapped the core when the rod was at the indicated zero position, the bottom of the cadmium section was approximately 1 in. above the bottom of the active reactor core. Nevertheless, excessive manual withdrawal of the central rod during assembly operations appears to have been the cause of the incident.

Conditions at the Time of the Incident. At the time of the incident, with the other four cross-shaped control rods at indicated zero position, it is estimated that the central rod would have had to be raised at least one foot more (to an indicated position of approximately 17 in.) for delayed criticality (see ref. 1, p. III-107).

A plausible reconstruction of the positions of the three men on duty at the time of the incident is given below:[6]

> "At the time of the explosion, the reactor crew was most probably engaged in reassembly of the number 9 control rod (the central control rod) assembly. To the Board, it appears quite plausible that the shift supervisor and the other regular member of the crew were located on top of the reactor vessel at the time of the explosion. The third member of the crew, who was a trainee, might have been partially over the reactor top or close to the edge of the reactor top. Based on medical evidence of injury, location of the men after the incident and knowledge of the individual capabilities and job assignments, it appears reasonable to hypothesize that the supervisor was in a crouched or squatting position which would be normal for manipulation of the C-clamp during the reassembly of the number 9 control rod drive. Similarly, it would appear reasonable to hypothesize that the regular crew member was located in a position for lifting the number 9 control rod assembly with the handling tool. There is no direct evidence to corroborate these hypothetical assertions."

Critique of Control-System Design Philosophy

Top- Versus Bottom-Mounted Control-Rod Drives. During the early search for causes of the accident, several possibilities were discussed

that illustrate the decreased accident potential of bottom-mounted control-rod drives. It was conjectured that hydrogen might have accumulated in the pressure vessel and exploded and that this explosion could have raised one or more control rods sufficiently to initiate a nuclear transient.

The mounting of control-rod drives on top of the SL-1 reactor was essentially mandatory. The technical characteristics limited the maximum height of the building to 50 ft above grade, and an airspace was to be present between the ground and the bottom of the reactor building (see ref. 2, App. V, pp. 257–265). In this compact design, there was no space for a subreactor area that would permit bottom mounting and, as necessary, maintenance of the control-rod drives.

The likelihood of a hydrogen buildup was extremely remote, in the state of the SL-1 assembly at the time of the incident, because of the relatively large clearances in the drive-mechanism components. A reactor design or configuration can be conceived, however, that would allow this sort of accident to occur with top-mounted drives. In the case of bottom-mounted drives, the lower grid structure usually serves as a positive stop of the control rods in their position of full reactivity control. Thus, a pressure transient would tend to hold the control rods in the configuration of maximum effectiveness.

In the SL-1 transient, the control-rod guide shrouds collapsed soon enough to trap the off-center control rods essentially in the position of full insertion. When the shield plugs were ejected from the vessel, the off-center rods were held by the shrouds, and breaks occurred at

weak points in the drive mechanisms. Otherwise the transient might have raised other control rods and might have been more energetic.

Although top mounting of control-rod drives has disadvantages, they must be weighed against advantages of access, when such exist, or the general advantage of greater assurance against accidental loss of water. The top mounting of rod drives in the SL-1 was not, in itself, a direct cause of the accident.

Assembly (Disassembly) of Drive to Control Rod. In retrospect, it is clear that a different method of assembly of the drive to the control rod should have been adopted; namely, one that did not require manual raising of the control rod. The sequence of assembly operations selected for the prototype SL-1 plant was not intended to be performed as a routine procedure at the ultimate sites for these plants. Nevertheless, less reliance should have been placed on the following of instructions and greater emphasis on a margin for human error.

In general, positioning one or more mechanical stops to limit the upward travel of the rod during assembly (or disassembly) does not provide enough margin for error. At some point during the procedure, the stops must be engaged and then they must be removed. Personnel performing the operation might become so dependent on the safeguard afforded by these stops that relaxed vigilance could lead to an accident because of an unrealized failure to actuate them.

It must be remembered that, when the rod is not attached to its drive mechanism, there is, normally, no indication of rod position. This ignorance of the position of the rod compounds the hazard, and exists whether the rods are moved manually or by other means. Therefore, such an

operation requires careful supervision and must not be considered routine. In this operation, as well as in others effecting reactivity changes, it would be useful to employ a check-list system.

All operating personnel should be made aware of known potential hazards of any of the operations that they might be called upon to perform. The written detailed instructions for these operations should emphasize the existence of these potential hazards. Such precautionary notes should be boldly evident.

Reactor Shutdown Margin. The magnitude of the reactor shutdown margin played no part in the chain of events leading to the SL-1 incident. The discussion that follows is intended to place shutdown margin in its proper perspective.

In the past, the concept of "reactor shutdown margin" has been defined, rather poorly, in terms of its significance to the hazards evaluation of a specific reactor design. Sometimes, the value quoted is the shutdown margin in the fresh reactor. This is defined to be the per cent reactivity by which the reactor is subcritical, with the normal control system fully effective and with the reactor otherwise in its condition of maximum reactivity. Typically, the condition of maximum reactivity is associated with the cold reactor with minimum void space and minimum xenon in the core. This concept of shutdown margin is too limited, for often the fresh reactor poses less hazard than one in which fission products have accumulated. More generally, the shutdown margin is the smallest value of reactivity that must be added to bring the reactor to delayed criticality, under all possible conditions, when the normal, adjustable reactivity-control system is exercising its maximum effect.

Even the more general concept of shutdown margin may be inadequate, however, as a criterion for assessing the safety of a proposed reactor. Bates[7] has published data for a number of power reactors to illustrate the marked variation in what was considered by various designers to be an acceptable or desirable shutdown margin. It should not be inferred, however, that a given reactor design with an actual shutdown margin of, say, 6% reactivity would have been unacceptable if the margin had been only 4%. Normal design practice requires that the calculated shutdown margin be conservative to allow for uncertainty in the calculation.

A margin of reactor control is required for a variety of reasons, and the magnitude of this margin is set accordingly. For example, the shutdown margin should be large enough so that there need be no great concern about the reactivity hazards of small errors in fuel loading. In the SL-1 reactor, the shutdown margin was in excess of 2% reactivity, and probably at least 3%, even without the six fixed cadmium blades installed in November 1960. The cold, fresh reactor was subcritical by approximately 3 1/2% with all five cross-shaped control rods at the indicated zero position. This margin was more than enough to override possible mistakes in fuel-loading changes, such as inadvertent loading of a fuel assembly without poison strips, misorientation of a fuel assembly, or addition of an extra fuel assembly at the periphery of the core. A considerable loss in shutdown margin may have occurred when, in September 1960, a fuel assembly near the core center was removed, together with what remained of its full-length and half-length burnable-poison strips. This assembly was replaced with a fresh, instrumented fuel assembly with a single, half-length poison strip.

Single-Rod (or Single-Mistake) Criticality. In the SL-1, each cross-shaped control rod was of a large span (14 in. of absorber). Moreover, the rods were so far apart that withdrawal of the central rod effectively left unprotected a portion of the core that was large in terms of the characteristic diffusion distance of thermal neutrons. As a result, the reactor was supercritical throughout its operation in the absence of the central control rod. Probably, the reactor was subcritical at all times in the absence of any one of the other control rods.

The ability to reach supercriticality with only the central rod withdrawn is a major factor in the explanation postulated for the nuclear excursion of the SL-1. It would be tempting to state categorically that no future reactor design should be accepted if, at some time in reactor life, and at certain conditions, withdrawal of a single component of the control system (e.g., a single control rod) would result in reactor supercriticality, even though the rest of the control system was fully effective. Certainly, it should be a prime goal of the design effort to avoid such a possibility. It may be, however, that in the avoidance of this undesirable situation, the reactor design would be affected adversely in other important respects. In such a case, the reactor designers must arrive at a decision as to the extent of a compromise of the conflicting goals. Additional control elements, playing the role of safety rods, might be introduced to increase the effective shutdown margin, when necessary. For example, in the case of a water-moderated reactor controlled by a system of control rods, boric-acid solution could be added to the reactor water during shutdown. Note that such extra control must not be considered to be an emergency measure; once additional control is

selected, it must operate as a normal, functional part of the reactor system.

Under some circumstances, where additions to the control system cannot be made, special safety devices, such as fuses, might be employed to ensure reactor shutdown.

A less restrictive condition that may be acceptable is that no single mistake in control-rod manipulation and no single malfunction of one component of the reactivity-control system would lead to an excursion that would damage the reactor or result in a serious hazard to personnel. For example, if, upon full withdrawal of a single control rod, the reactor would be below prompt critical, a relatively slow rise in the level of the neutron flux would result. The reactor should be capable of compensating for this excess reactivity without a severe transient. Preferably, such compensation should be inherent, but, here again, the use of special safety devices, such as fuses, or of very carefully controlled operating and shutdown procedures may be sufficient in unusual cases. In any event, the reactor design should be such that, whether or not the reactor is damaged, personnel in the area would not suffer serious injury. For some types of reactor, this less restrictive criterion might be satisfied, even if the net uncompensated reactivity was more than one dollar.

Soluble Poison System. Current technology supports the use of boric acid solution as a routine method of providing additional control in a water-cooled reactor, such as the SL-1. At the time of design of the SL-1, however, soluble poisons were considered unproved for routine use at the specified operating conditions. The practice of using solid

absorbers is firmly entrenched, and the prejudice in favor of using only "tried and true" methods dies hard.

Although soluble poison might now be considered to be a standard control method, its use still requires certain precautionary evaluations. For example, it cannot be used in a reactor core where moderator expansion causes more reactivity to be gained by loss of poison from the core than is lost by decreased moderation. Thus, a soluble poison is inappropriate in a substantially overmoderated core. Additionally, in some reactors, under certain circumstances, a soluble poison may present a hazard of another sort. If poison solution is added to a system containing a limited volume of water, the resulting increase in moderator volume might well overcompensate the reactivity effect of the poison. An extreme case would occur with the addition of poison to the system outside the core under circumstances in which it would not be well mixed with core water. These reservations about the use of soluble poison are most pertinent to the case of a cold, shutdown reactor, since in this condition, the water level is likely to be low, the mixing poor, and the moderator density high.

In retrospect, it was proper that boric-acid addition was not designed as a fast-acting emergency shutdown measure in the SL-1. Such emergency shutdown systems must be capable of reliable and rapid injection of poison into the core, and experience at Argonne has indicated that soluble-poison systems are not amenable to these criteria.

A redesign of the SL-1 at this time would feature several changes in the control scheme. Certainly, one change would incorporate a soluble-poison system, in particular, where conditions precluded provision for

shutdown with one control rod withdrawn. The soluble-poison system would be designed for routine operation, at least during shutdown. Moreover, a minimum of one control rod would be designated to function exclusively as a safety rod during reactor shutdown.[8]

Conclusions

Two design decisions appear to have made the SL-1 incident possible: (1) the decision to accept a design in which the reactor could be made supercritical by the removal of a single control element, and (2) the decision to design control-rod drive mechanisms that necessitated manual manipulation of the rods to effect assembly or disassembly. In both cases, the designers were aware of the difficulties and took additional precautions to reduce any potential hazards. Nevertheless the potential accident seems to have occurred.

Regardless of design, the potential exists for achieving supercriticality by manipulation of a single control element. For example, even with proper design for multiple-rod startup, an error in reactor loading can negate the design intent. It might be reasonable to suggest that possible loading errors should define the shutdown margin; however, unless operating procedures are rigidly enforced and reactivity is checked continuously, a system which is too reactive could, at some time, be assembled.

It may also be an illusion to insist that manual manipulations of control rods cannot be allowed at any time. For example, if the replacement of rods is specified to require special, slow-moving machinery, it

is necessary to make sure operationally that alternate means of rapid removal are not attempted.

Thus, although the standard conditions of design should provide for continued shutdown in the absence of any one control rod and servicing of control mechanisms with devices of limited speed and capabilities, the vital components of reactor safety remain vigilance and caution during all operations.

The discussions that followed the SL-1 incident support the view that bottom mounting of control-rod drives is more conducive to safety in operation and in maintenance, if bottom mounting is feasible. The advantages of the concept are not sufficient to merit listing it as a criterion in the design of water-cooled reactors, but bottom mounting is recommended as a first approach.

Similarly, discussions of the SL-1 incident lead to the general conclusion that soluble-poison control, when applicable, is a preferred method of effecting emergency reactor shutdown, even though subsequent removal of the poison is intrinsically a slower and more complex process than is the removal of a rod. Design conservatism resulting in a reluctance to apply soluble-poison control may well be a poor approach to reactor safety.

The lack of an emergency shutdown system was a contributing factor to the SL-1 incident. In the absence of normal control devices, fuses can be considered, but these are not well regarded because of their generally random characteristics. It appears that fuses are not the answer to safety problems during operation, but properly designed fuses, with the sole function of maintaining shutdown conditions, are better than

nothing. The applicability of a fuse depends on the availability of a unit with appropriate negative reactivity-time characteristics for contemplated accidents; these characteristics are not always available. Moreover, if a negative reactivity can be inserted by a fuse, it is preferable to insert this negative reactivity beforehand in a more controlled manner. There may be some situations, however, in which normal reactivity reduction in a shutdown core could lead to awkward startup circumstances; fuse mechanisms for emergencies might then be appropriate.

The SL-1 accident is illustrative of inadequate communication between reactor designer and reactor operator. The designers must bear the burden of complete reporting on the design and the concepts that motivated the design approach. The detailed background of the SL-1 design was not spelled out by complete design education of the operators or complete reporting of the design concepts.

The current emphasis in reactor safeguard analyses is on design. One lesson learned from the SL-1 incident is that there should be greater emphasis on reactor operations and procedures because of their significance to reactor safety and the increased light they shed on the safety aspects of design approaches.

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IN-CORE TEMPERATURE MEASUREMENT

Nuclear reactors are large heat-generating machines in which temperature is undoubtedly the most important nonnuclear measurement required for safe and efficient operation. Starting with the first graphite reactor, thermocouples have been used extensively for the required temperature measurements. After approximately 20 years of experience, thermocouples have emerged as the only proven sensor that can be located in-core and withstand the reactor environment for long periods of time. Chromel vs Alumel has been the most widely used combination of materials and thus constitutes the thermocouple with the highest confidence level. Thermocouples are not ideal, but for temperatures below 1500°F they meet the following criteria for an in-core sensor: simple, rugged construction; reasonable accuracy for long periods of time; small size, approaching a point measurement; and insensitivity to nuclear radiation.

As new reactor concepts emerge and reactor temperatures are pushed higher and higher, the time has come to reassess in-core temperature sensors. At temperatures in excess of 1500°F, Chromel-Alumel thermocouples show an appreciable drift with time, and for temperatures in excess of 1800°F they no longer have either an acceptable life or accuracy. Considerable research time and money is presently being expended on the development of new types of thermocouples that utilize noble or refractory metals.

The advisability of an extensive effort to develop high-temperature thermoelectric-type sensors for long-term in-core application is questionable for the following reasons:

1. The thermoelectric output of a metal is a complex phenomenon that is affected by changes in trace quantities of many alloying elements. Chemical activity increases with temperature and increases the probability of changes in composition and thermoelectric drift.

2. Insulators such as beryllia, alumina, etc., undergo drastic changes in electrical characteristics at elevated temperatures. The resistivity decreases exponentially with temperature, and the dielectric constant increases exponentially. The resistivity of beryllia,[9-14] which is considered to be one of the best insulators, decreases by 10^3 between 1450 and 2450°F. Trace element impurities in insulators are known to seriously affect resistivity, and it is anticipated that they will also produce semiconductor effects.

3. Most refractory metals that have been proposed as thermocouples undergo nuclear reactions when used in-core. Browning and Miller[15] have calculated these composition changes and have determined that there are appreciable changes at a thermal-neutron flux of 1×10^{14} neutrons/cm^2·sec after 1 year. A composition change does not necessarily mean a change in thermal emf, but at the present state of the art the only real proof of accuracy will be long-term in-core tests.

4. Most refractory metals must be maintained in an inert atmosphere. This not only seriously limits their flexibility but makes them of questionable reliability for reactor operation and safety applications.

The development of high-temperature thermocouples should be continued, but it seem advisable that considerable effort should also be expended to develop temperature sensors that utilize other phenomena. Such sensors should be as independent of effects produced by their

materials of construction as possible, able to withstand the reactor environment, and have a simple and predictable relationship with temperature.

Several types of sensors have been proposed for study, and three have been discussed in the literature on which considerable research time has been expended and which have good possibilities.

Acoustic Thermometer

The acoustic thermometer,[16] as proposed, makes use of the relationship of the velocity of a low-frequency sound wave in a gas to the absolute temperature of the gas. To the first order, velocity depends on molecular weight, absolute temperature, and the ratio of specific heats. The temperature-measuring apparatus consists of an acoustic cavity (1/2 in. in diameter and 10 in. long) coupled by a small-diameter tube (1/16 to 1/4 in.) to a remotely located (10 to 20 ft) transceiver and the associated electronic equipment. The resonant frequency, and thus the temperature of the gas in the cavity, is measured by applying short bursts of sound energy to the transceiver and measuring the reflected echo signal. The advantages of such a system should be freedom to select any material that will contain a gas and insensitivity to nuclear radiation. The disadvantages of such a system would be the large sensing element relative to a thermocouple and the limited allowable distance between the cavity and the transceiver.

Pneumatic Thermometer

The pneumatic thermometer utilizes a gas stream flowing through two nozzles in series, both operating with pressure drops in excess of that

required for critical flow. The first nozzle is located remotely, at a convenient location, where its temperature can be measured and controlled. Pressure is measured upstream of each nozzle, and the temperature of the gas at the second nozzle is derived from the relation

$$T_2 = K \left(\frac{P_2}{P_1} \right)^2 T_1 ,$$

where T_2 and P_2, and T_1 and P_1, are the temperature and pressure at the inlet to the second and first nozzles, respectively; the symbol K represents the system constant.

Such a system is presently being installed in the Experimental Gas-Cooled Reactor (EGCR), and the details of the system were discussed in an earlier issue of Nuclear Safety.[17] The piping and valves of an existing burst-slug detection system were used for the EGCR pneumatic thermometer system, and as a result the system has large tubes, nozzles, and gas flow rates. There is reason to believe that other configurations can be used to reduce the size to that of the commonly used 1/8-in. sheathed thermocouple. The advantages of such a system should be freedom to select a wide variety of materials, insensitivity to nuclear radiation, no practical limit to the distance between the two nozzles, and ease of recalibrating and checking of the nozzles during reactor down periods. The disadvantages of such a system would be its slow response and fairly large sensing area.

Capacitance Temperature Sensor

The General Electric Company, as part of its contract with the AEC for the development of reactor instrumentation and controls, has been

investigating a capacitance temperature sensor.[9-14] This work has not progressed to a final design for a sensor, but the significance of the work done to date is in the potentialities of a ceramic sensor, such as beryllia or alumina. The sensor used for development work is a rod, 1 in. long and 0.238 in. in diameter, with parallel platinum plates inserted into slots in the rod. Readout is obtained by operating the sensors as part of a radio-frequency oscillator, and temperature is indicated as a function of frequency.

Data presented to date for beryllia show very large changes in both resistivity and capacitance with temperature that can be represented by the following relationships:

$$R = \frac{K_1}{T^{17}} ,$$

and

$$C = K_2 T^{17} ,$$

where R is resistivity, K_1 is a constant, T is temperature, C is capacitance, and K_2 is another constant. Much more development work is required to determine whether these electrical characteristics can be utilized for a durable and accurate temperature sensor, but a device offering such high sensitivity deserves careful consideration. (C. A. Mossman)

THE ROLE OF THE LOG N PERIOD METER IN REACTOR PROTECTION

The Log N period meter is probably the most misused and least understood instrument in a reactor system. It is a relatively complex instrument that utilizes components and techniques with inherently low reliability, and it is difficult to monitor or to test for proper operation.

The input stages of most Log N amplifiers consist of either a diode operated at low heater voltage, followed by an electrometer, or a special pentode. All these exhibit drift and temperature effects and sometimes suffer permanent changes in their electrical characteristics when subjected to shocks, such as connecting or disconnecting the input signal from an ionization chamber. Some circuits have been built using solid-state diodes as logarithmic elements, but the extreme temperature sensitivity and hysteresis exhibited by these devices renders them generally inferior to their vacuum-tube counterparts.

Misadjustment of the compensated ionization chamber, normally used as a signal source for the Log N, to a condition of overcompensation can cause a temporary paralysis of the input, which renders the instrument useless for short-period protection because of excessive delay. This same effect of overcompensation sometimes causes false scrams during startups on ordinary periods because of the instrument suddenly recovering from its paralyzed state. A circuit modification can be made to correct this difficulty, but few Log N amplifiers now in service contain this modification. This effect and other difficulties, such as extraneous noise pickup and poor combinations of scram set points and smoothing-time constants, have been responsible for a large number of false scrams;

that is, a number far out of proportion to the enhancement of safety provided by the Log N period meter. It is interesting to note that there have been virtually no significant improvements in this instrument since it was invented in 1948,[18] in spite of its widespread use and commercial availability.

In spite of its poor reputation, the period meter can be an important adjunct to reactor safety systems when properly applied. The poor reliability and difficulty of performing meaningful on-line tests, as mentioned above, usually rules out the use of the period meter as a primary safety instrument. The role of the period meter then becomes one of augmentation of the safety system to reduce the severity or magnitude of excursions, recognizing that the failure of the period safety to function will result only in exercise of the level safeties, which must work. In this secondary role, the period safety should not be permitted to interfere with operation by having its trip point set too close to normal operating periods, either transient or stable. If the trip point is set for a period substantially shorter (1 to 3 sec, for example) than the normal operating periods, the very existence of such a period is prima facia evidence that the control system has failed and a power excursion is in progress. If the scram point is set closer to operating periods, it becomes necessary to increase integrating time constants in the period circuitry to reduce the false scram rate from random and spurious noise on the period signal. This results in increased response time to short periods and generally inferior performance.

The theoretical time response of a typical Log N period meter (ORNL-Q-915), which is in widespread use, has been described in detail

by Ditto[19] and verified experimentally by a series of experiments at SPERT.[20] The calculated and experimental data are compared in Fig. III-3. In this set of curves, the chamber signal current at the instant of scram, rather than time, has been shown as the significant variable. It should be noted that chamber current and reactor power are directly related in these curves so that 5×10^{-5} amp corresponds to full power. The calculations and experiments show that the period system can initiate a period scram at a current three decades below the level scram set point when the period is 3 msec and the initial current is below 10^{-11} amp. The scram occurs at even lower power levels for longer periods. Ditto has shown also that relatively large steady currents (10^{-8} amp dc) such as might occur with undercompensated ionization chambers have very little effect on the response to short periods. For short periods the current (i.e., reactor power for a fixed chamber location) at which scram occurs for a given period is approximately directly proportional to the shunt capacitance across the log diode (typically, 10^{-9} farads, including cable and circuit) and inversely proportional to the period scram set point. As the initial neutron-induced current is increased, the current at which period scram is initiated increases so that the period meter affords less anticipation of a level scram for transients initiated at high currents.

As Ditto points out, the ultimate effect of the action of the period scram is to provide additional time for the safety mechanisms to operate by setting them in motion several decades below full power. Obviously the advantage is lost if the excursion starts near full power or if the response of the safety mechanism is long compared with the anticipation

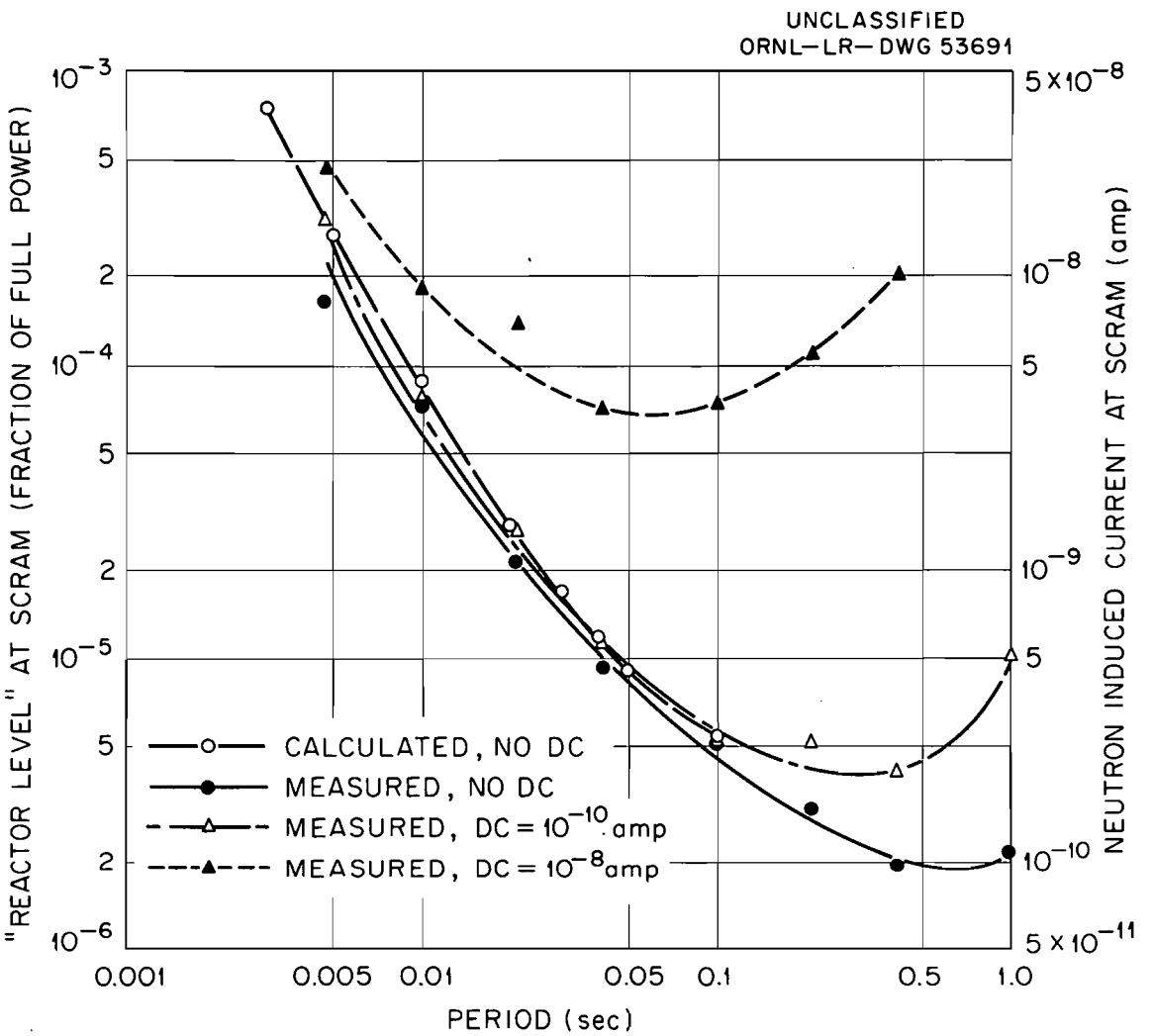

Fig. III-3. Effect of Direct Current Component on Period Scram Response.

provided by the period meter. Since the level safety and period safety actions utilize the same control rod release mechanisms, a slow level safety invariably causes a slow period safety. Thus it is incorrect to imply that the use of a period scram allows the use of slow level safety systems. Quite the opposite, if a period scram is to be of value, the safety system delays must be short.

A short period scram has very little value for excursions initiated at high power levels. Most reactors have negative power coefficients of sufficient magnitude to make it difficult to get on short periods at high powers, but even if a short period could be established, it is unlikely that a period scram would occur much, if any, before a level scram because of the relative closeness of the level scram point when operating at or near full power.

Some reactors have severe limitations on permissible rate of change of temperature or power in the sensible heating range for reasons of thermal shock. In such a reactor, it might be tempting to use a period meter with a trip level set for relative long periods comensurate with allowable rates of change of temperature. This would be a poor choice, however, for several reasons. Such a meter would require long smoothing time constants, as described above, in order to reduce the probability of false scrams from noise and safe transient periods, with a resultant slower response to unsafe periods. Also, most reactors have a linear rather than an exponential character in the sensible heating range because of negative power coefficients. A better safety parameter than period, then, is the linear rate of change of power, which can be derived simply and reliably from linear power level signals.

A recent report by Fowler[21] describes a new reactor period meter with which great effort was made to increase the speed of response at high power levels. The trip level is set at a 20 sec doubling time, which corresponds to a period of about 28 sec. The reviewer is not familiar with reactors that require this unique condition of high-speed response coupled with a long period trip point, but this seems to be a combination guaranteed to keep the reactor scrammed most of the time with a negligible improvement in protection. This "improved" response at high powers was apparently obtained at the expense of slower response at lower power where a fast response might reasonably be expected to offer some advantage. A rate of change circuit might better serve the purpose and allow the period meter to be tailored for better response at low powers. Fowler also includes a negative period scram on his period meter for the expressed purpose of providing protection against an accident in which a control arm falls through the reactor and out the bottom, thus giving a negative period followed by a short positive period. This has the effect of substituting a period meter for a proper control rod design with adequate mechanical stops ("bottom of the pile") and is grossly unacceptable in any safety system. It is difficult to conceive of a situation in which a negative period scram can be of value. It is most often proposed as a crutch to, in some way, justify the inadequacy of some other component of the system.

Perhaps the most significant reason for misuse or misunderstanding of the period meter is the lack of a universally accepted way of expressing the time response of the instrument. Most manufacturers either ignore the problem completely or quote a meaningless time constant. An excellent

way of describing the response is by a family of curves, such as in Fig. III-3, which describe the response of the ORNL Q-915 Log N period meter for short periods initiated at 10^{-11} amp or less in terms of current or power rise before scram occurs. To complete the family, curves should be added for higher initial currents. It can be argued that reactor excursions are seldom characterized by constant periods, but by selecting the shortest predicted period, a pessimistic value for the response can be obtained.

A direct measurement of the response of a period meter can be made with reasonable accuracy with very simple equipment. An exponential current generator can be assembled with analog computer components that will cover a three-decade range. This generator, together with a calibrated oscilloscope, is all that is needed to measure the response over a reasonable range of periods. (J. L. Anderson)

REACTOR BEHAVIOR ANALYSIS USING ANALOG COMPUTERS

The analog computer has proved to be a very useful device for studying the dynamic behavior of physical systems while these systems are in the design stage and prior to their actual operation. From these studies, many design changes are initiated that may make an inoperable system operable, an unsafe system safe, or an uncontrollable system controllable. The optimum mode of operation of a system for achieving the desired goals in a safe manner can also be determined by use of the analog computer.

In the current literature there are many articles dealing with the use of analog computers in systems analysis work. One of these articles will be briefly reviewed here. The primary aim of this review is to evaluate the usefulness of the information to others in the nuclear reactor field, particularly to systems analysts who use analog computers in their work. The report covers work done by the Holley Carburetor Company for the Power Reactor Development Company.[22] The work consisted of a systems analysis of the Enrico Fermi atomic power plant, and the building of an operator-training simulator.

At the beginning of the program, it was not known whether the plant could best be simulated on a digital or an analog computer. Both approaches were used simultaneously so that the most promising one could be employed without loss of valuable time. The digital computer was in contention to the point where simulation of the "once-through" steam generator transient behavior was attempted. It was found that this computation would have required an unreasonable amount of computer time. In order to maintain satisfactory accuracy, a 5-min real-time interval

would have required 2000 hr of machine time. The simulator ultimately selected consisted of a general-purpose analog computer (Beckman/Berkeley 1100 Series), an assortment of passive electrical components, and some electromechanical components. The digital computer was used to check the analog model and to do steady-state calculations.

The Enrico Fermi atomic power plant consists of a nuclear reactor, an intermediate heat exchanger where the primary and secondary sodium coolants exchange heat, a steam generator where the secondary sodium coolant and water exchange heat to produce steam, and a steam-driven turbine for producing electricity. In addition, of course, there are numerous pumps, valves, and associated equipment. There are actually three complete heat transfer loops; however, only one loop was simulated because it was assumed that all three loops were identical. In the simulation, general-purpose electronic-analog-computer equipment had first priority and was used whenever feasible and economical; otherwise passive elements and electromechanical devices were used.

The reactor and the mixing effects of the plenums at the inlet and outlet of the various components were simulated with circuitry in the analog computer based on differential equations. The transport-delay phenomenon and the intermediate heat exchanger were simulated by specially designed electromechanical equipment. The steam generator was simulated by a combination of special-purpose equipment and electronic circuitry in the computer.

To simulate transport delay, the so-called "electrical bucket brigade" was used. Each of these devices consisted of a rotating wheel on which was mounted a bank of capacitors. The wheel was made of plastic

and had 100 brass commutator bars mounted on its periphery. These bars were wired to 100 polystryene capacitors, also mounted on the wheel. The wheel was driven by a velocity servo through a precision gear train. The speed of the wheel was varied to simulate changes in the flow of the fluid in the physical system being simulated. The voltages across these capacitors were proportional to the temperatures.

The counter-flow intermediate heat exchanger was simulated using two wheels rotating in opposite directions, with stationary "T" electrical networks connecting the commutator bars on one wheel to the bars on the other wheel through sliding contacts. In this way, transfer of electrical charge was analogous to transfer of heat.

The steam generator simulation was accomplished by using a capacitor wheel on the hot side, with stationary capacitors and resistors for the tube metal and water side. A rather complicated computation, too lengthy to be discussed here, was carried out to simulate proper heat transfer from sodium to water.

Whenever it was necessary to determine a capacitor voltage, the voltage was read into each plenum circuit through a cathode follower, which had essentially infinite input impedance, in order to sample the voltage without discharging the capacitor.

The circuit that simulated the reactor neutron kinetics consisted of numerous passive electrical elements plus two analog-computer amplifiers and one multiplier. It is worth noting that through judicious use of passive electrical elements and electromechanical devices, the required general-purpose analog-computer equipment was limited to the following:

Item	Quantity
Operational amplifiers	130
Electronic multipliers	18
Electronic function generators	16
Amplifier-driven relays	16
Coefficient potentiometers	110
Silicon diodes	20

The simulator was used to investigate the transient response of the system to various perturbations and for studying various accidents. It was also used in the development of a control system for the plant. With the addition of a control console, it was used to train plant operators.

In the reviewer's opinion, the very comprehensive report reviewed here would be of great value to anyone interested in simulating a nuclear power plant. (O. W. Burke)

THE ROLE OF THE NEUTRON SOURCE IN REACTOR SAFETY

Startup Kinetics

Students confronted for the first time with the neutron chain reaction invariably ask whence the "first" neutron comes, and a discussion of neutron sources ensues. Practical sources are then described, and the emission rate required of the source is related to the effective sensitivity of the counting-channel sensors and to the subcritical multiplication of the shutdown reactor. The inference is drawn naturally that the source is provided to make the counters count.

This inference is untrue. The importance of the source can best be understood with regard to the probabilistic, or statistical, nature of the chain reaction. The multiplication, k, is the average number of neutrons released per neutron lost. The variables that appear in the reactor kinetic equations as they are usually written are expectation values of physical variables: neutron population and delayed-neutron precursor populations. The "constants" are averages characterizing probabilities such as leakage, capture, and radioactive decay. For the populations usually encountered, the law of large numbers holds very well; an enormous body of experimental data confirms the correctness of the kinetic equations.

When the neutron population of a reactor is very small, on the other hand, the variables of interest are subject to nonnegligible fluctuations. For a population of zero neutrons, to take an extreme example, the value of k will have no effect whatever on the behavior of the reactor, which will of course remain neutronless. It is clear that if a neutron should

happen along in such a situation, and if this neutron should be so fortunate as to cause a fission, the sequence of subsequent events would depend on occurrences whose probabilities could be predicted only a priori. In particular, the k of the reactor at the time of the fission would have an important effect.

These conjectures have been substantiated by the results of experiments conducted with Godiva,[23] HPRR,[24] and BSR-II.[25] In each case, a reactor with a very low source level was rapidly brought to a condition of $k \gg 1$, so that the kinetic equations predicted a very short stable period, that is, a period in the microsecond range for Godiva and HPRR and in the millisecond range for BSR-II. The predicted short periods were, in fact, observed; the measured periods agreed with the theoretical predictions, and the validity of the kinetic equations was confirmed, as usual. A very peculiar feature of the excursions in all cases, however, was a long delay between the time at which $k > 1$ and the beginning of the short-period rise in the neutron level. The starts of millisecond- or microsecond-period rises were delayed as much as 1 sec. The explanation of this delay is simply the very small neutron source which was present.

Most reactors are not started by increasing k above prompt critical in a few milliseconds to induce a short period. As described by Newson,[26] the maximum rod speeds and safety-system response are selected so that the excursion caused by continuous rod withdrawal at maximum speed is tolerable. A slower rod speed must often be provided so that the operator and the control system can cope with the changes in k and an orderly startup will result. Calculations of the behavior of the reactor under these conditions are made using the kinetic equations.

The role of the source in normal reactor startup is then clear: the source must be of sufficient strength that the reactor behavior will be governed by the kinetic equations, with acceptable fluctuations. The control system will function, and the safety system will provide protection, in accordance with the predictions of the known behavior of the reactor.

Source-Range Instruments

The primary function of the source-range instrumentation is to confirm that the necessary source is indeed present. In order that this be accomplished reliably, the instrumentation must detect (usually, count) neutrons from the source, and not electrical noise from welders, fluorescent lamps, telephone dials, and the like. It is unfortunate that many source-range instruments now in use do not meet this minimum criterion.* Attempts to improve this situation by installing several identical counting channels do not usually alleviate the situation, since the difficulty lies in design and installation errors in the instrumentation. Two counting instruments have recently been described[27,28] in which design features have been incorporated to minimize troubles from electrical noise pickup.

Since the source-range instrumentation often supplies the only information the operator can use while starting the reactor, it is not surprising that much effort has been expended in making the range, time response, etc., of the instruments convenient for reactor operation and even for reactor "safety" — scram, that is. The point to be made here is that the essential function of these instruments — confirmation of the presence

*Based on communications between the reviewer and numerous reactor operators.

of the source — must not be compromised by other uses, however fortuitously convenient they may appear. Of especial concern to the reviewer is a proposal by Barrow and Maitland[29] who analyzed the transient response of a period computer that used a signal derived from a logarithmic count-rate meter. They concluded from their calculations that "A much too high growth factor [inverse period] will ... be indicated by the period meter, and 'scram' may be called for when all that was needed was patience. To avoid this type of response when operating with a reactor, a bias current, equivalent to two to four counts per second, should be kept on the diode." It is evident that this proposed bias is a false indication of a source in the reactor on what is, after all, the source-range channel; the reviewer believes that such a circuit should not be used.

Source Level

Although fission-product radiation normally provides a "built-in" source that is more than adequate for restarting a reactor, the first startup of a power or mobile reactor may present a serious problem. Logistic difficulties may make transporting a radioactive source unattractive; the configuration may make access impossible. Theoretical studies[30-33] have been concerned with the divergence of reactor startups at low source levels that proceed with slow rates of change of multiplication. For each reactor configuration, the usual methods of analysis are used to establish the maximum reactivity at some reactor power above which thermal effects and safety-system response determine the course of the excursion. This reactivity is directly related to the reactivity insertion rate and to the "effective source" at the threshold reactor power

above which the fluctuations no longer control the behavior. Finally, a stochastic theory is used to estimate the probability that the effective source will be less than the minimum required to ensure that the excursion is tolerable. Some representative results of such calculations are shown in Table III-1. It is clear that the approximate probabilities (which are only intended as order-of-magnitude estimates) are very sensitive to source level.

The natural source level to be expected in some power reactors of interest has been calculated[34,35] and measured.[35,36] In general, the measured values are lower than the calculated values by a factor of 2 to 5. It would seem advisable therefore to refine the calculations or make careful measurements for reactors in which this is an important parameter.

It is noteworthy that boron added to an enriched alloy fuel as burnable poison contributes substantially to the source level as a result of high-energy alpha particles from U^{234} reacting with B^{10}. The startup times for a reactor with initial reactivity several dollars subcritical is 1/2 to 2 hr for the examples given, which is a long time but is not intolerable for a first startup. In conclusion, "sourceless" startup may be attractive, if the source is not too low. (S. H. Hanauer).

Table III-1. PROBABILITY OF UNACCEPTABLE TRANSIENTS

Case	1[a]	2[a]	3[b]	4[b]
Effective source	50 neutrons/sec	850 neutrons/sec	100 neutrons/sec	200 neutrons/sec
Rate of increase in multiplication	5×10^{-5} Δk/sec	5×10^{-5} Δk/sec	0.1 ¢/sec ($\sim 7 \times 10^{-6}$ Δk/sec)[c]	0.1 ¢/sec ($\sim 7 \times 10^{-6}$ Δk/sec)[c]
Probability of exceeding tolerable excursion	3×10^{-3}	$<10^{-8}$	10^{-3}	10^{-6}

[a]Unpublished Westinghouse Electric Corporation data.

[b]From ref. 33.

[c]Estimated by editor.

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IV. PLANT SAFETY FEATURES

IN-PLACE TESTING OF HIGH-EFFICIENCY FILTERS

The application and importance of high-efficiency filters in atomic energy installations for the removal of submicron radioactive particulate matter is well known.[1] In most exhaust systems the high-efficiency filter is the primary means of minimizing atmospheric contamination during normal operation or during the accidental release of radioactive particulate material. These filters, if in good condition, will remove 0.3-μ particles from an air stream with an efficiency of 99.97% or better. To achieve such efficiency in an exhaust system employing one or more filters, however, all factors that can contribute to leakage through or around the filters must be considered.

Shortly after high-efficiency filters became commercially available, purchasers recognized the need for routine inspection to assure, insofar as possible, the receipt of undamaged filters that met specification requirements with respect to efficiency. Because of the importance of these filters in the nuclear field, a filter inspection and testing service was established by the United States Atomic Energy Commission that was available on a voluntary basis to participants in AEC programs.[1,2] While this resulted in an improvement in the situation, shipping and handling damage continued to be a problem.

In order to assure adequate filters for its own use, the Oak Ridge National Laboratory checks each filter for efficiency[3] and sends to stock only those that comply with the specifications. While this is another step in the effort to prevent atmospheric contamination, it is obvious that the efficiency of a filter in its carton is relatively unimportant, except as an interim measure in an over-all program.

The efficiency of an installation or system not only depends on the individual filters, but also on the system as a whole; for example, filters must be properly installed to prevent leakage at the gaskets, and other leakage paths must be eliminated.[4] In the final analysis, an adequate system can only be attained by measuring the efficiency of the actual system with the operational filters in place. Nevertheless, most users of high-efficiency filters have been hesitant about undertaking a program of in-place testing.

In-Place Testing Procedure

The technique used for testing individual filters has been successfully adapted at ORNL to in-place testing of filter systems. The procedure used consists of discharging the polydisperse aerosol of dioctylphthalate (DOP), produced by atomization with compressed air, into any convenient air intake ahead of the filter bank. The concentration of the unfiltered smoke (aerosol) is then measured by drawing a sample of the smoke from a point ahead of the filter bank and passing it through a forward "light"-scattering photometer. The concentration of the smoke 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). The efficiency of the filter bank is then calculated from the unfiltered and filtered concentration measurements.

In order to conduct the test described above, there must be a sufficient length of duct between the point of introduction of the aerosol and the filter bank to induce thorough mixing of the aersol and the gas (usually air) in the system. When the length of duct between the point

of smoke injection and the filter bank is too short to induce good mixing, however, a reliable in-place test can be accomplished by modifying the testing procedure. The aerosol is introduced in the same manner, but only the filtered sample is removed from the system. The equivalent upstream concentration is measured indirectly by installing smoke generators downstream of the filters and sampling further downstream before or after the blower. This indirect method of obtaining the concentration of the unfiltered aerosol-air mixture is reliable if care is taken to maintain the air flow constant during the testing period. Consistency of the air flow through the system can easily be checked by the use of a pitot tube inserted in the duct. The output of an aerosol generator is dependent on the pressure of the compressed air at the generators and, to a lesser degree, on the liquid level of the DOP. Obviously, the output of the generator must not be altered during this type of test.

An alternate method makes use of the fact that the output of the smoke generator is dependent upon the air pressure at the generator and the level of the DOP. Each smoke generator can be calibrated in a system of known air flow, and an output versus pressure curve can be drawn for various liquid levels. Then, if the air flow in the system under test can be accurately measured, the upstream smoke concentration can be calculated by referring to the generator calibration curve. In such tests, the only sample needed for a particulate concentration measurement is one withdrawn downstream of the filter bank.

The modified techniques that require no upstream sample are particularly useful when the system upstream of the filter bank is highly contaminated, because they minimize the possibility of contaminating personnel

and equipment during testing. It is not always possible, however, to avoid contaminating the photometer during in-place testing, and thus the usual precautions for handling contaminated equipment must be taken. It is recommended, therefore, that at least one instrument be appropriately tagged and used only for testing contaminated installations, since this will keep the number of "in service" hours to a minimum and help to prolong the intervals between repairs. While the chamber of the photometer in contact with the sample can be decontaminated, to a certain degree, each time this is done the chamber must be carefully reblackened to eliminate internal scattering of stray light.

When a contaminated sample is withdrawn from the duct through the photometer, it should either be returned to the duct or passed through a high-efficiency filter before being released to the atmosphere. A breathing mask canister of the M-11 type installed between the photometer and the pump will serve in the latter case and will also minimize contamination of the pump.

The generators used at ORNL are a modified version of the aerosol generator described by J. K. Thompson.[5] Six atomizer nozzles, instead of one, are installed in a 5-gal can, with the nozzles located about 1 1/2 in. above the bottom. The jet impactors are omitted to increase aerosol production, although this has the effect of increasing the diameter of the average particle from about 0.3 to 0.7 μ. Little significance is attributed to the change in particle size for in-place testing applications.

One generator will supply sufficient smoke for testing systems up to 2000 to 4000 cfm, depending upon the type of photometer used; the

4000-cfm rate is needed for instruments with a gain control. Larger installations can be tested simply by manifolding two or more generators together.

Testing Followup

When an in-place test produces an unsatisfactory efficiency, corrective action must be taken. The filters can all be replaced, of course, but this is hardly an economical approach to the problem, and it may not produce any appreciable improvement in the conditions. Ideally, the source of the trouble can be found by probing the downstream side of the filter bank and filter mounting frame with the probe connected directly to the photometer. Leakage paths can be readily detected by observing the behavior of the pointer on the amplifier. In new systems it is not unusual to find gross leakage paths in the filter housing; intermittent fillet welds may have been used instead of continuous seal welds. Finally, after all leakage paths in the structure are permanently sealed, preferably by welding, the gasket areas and the faces of the filters should be probed. If the filters were installed without the benefit of experienced supervision, leakage will probably be found through the gaskets, as well as through holes in the filter medium made by careless or inexperienced workmen.

Since probing is sometimes difficult to accomplish and at the very best time-consuming, craftsmen should be given training in the care and handling of high-efficiency filters, and the complete installation procedure should be supervised by a responsible individual. The AEC has issued an excellent guide[6] covering the inspection, testing, and

installation of filters which is useful in training and in developing installation procedures. Installation is considered as commencing with the opening of the filter carton and ending with the closing of all access doors into the system. The procedures described pay dividends, since, if all necessary precautions are taken and unsatisfactory efficiency is found by the in-place test, the difficulty can usually be found and corrected with a minimum of effort.

Good filters, properly mounted in a structurally sound enclosure, with the system efficiency verified by an in-place aerosol test, constitute the essentials of a realistic program for the containment of radioactive particulate matter. The program does not stop here, however; routine in-place testing is important for maintaining the integrity of operational systems.

ORNL Experience

The results of numerous in-place tests at ORNL are presented in Table IV-1. The data, which do not include efficiency measurements made after repairs, emphasize the need for in-place testing, particularly for multifilter installations. Based on the data of Table IV-1 and experience in general, the following conclusions may be drawn:

1. It is relatively easy to achieve a system efficiency of 99.9% or better in small single- and double-filter installations.

2. Filters in chemical laboratories may deteriorate rapidly and should thus be checked frequently in place.

3. Relatively few existing, large, multifilter installations have an efficiency of 99.9% or better when initially tested.

Table IV-1 RESULTS OF INITIAL TESTS ON ORNL FILTER SYSTEMS

Type A System: 1- and 2-filter systems primarily of the chemical laboratory exhaust-hood type in service 4 months or less
Type B System: Same as type A except in service 14 months or less
Type C System: Larger multifilter installations of various types in service for various periods of time

	Type A Systems	Type B Systems	Type C Systems
Number of systems with efficiency of 99.97% or higher on initial test	129	122	5
Number of systems with efficiency of less than 99.97%	10	55	23
Total systems	139	177	28
Percentage of total systems with efficiency of 99.97% or higher on initial test	93	69	18

4. New systems of all types have a reasonably good chance of passing an in-place test if close attention is given to details of design and fabrication and the filters are installed under competent supervision.

5. The in-place test for determining system efficiency can be applied to most installations.

The cost of high-efficiency filters was discussed in the previous issue of Nuclear Safety.[7] The equipment for conducting in-place tests is not especially expensive, but the expense does depend on the type of photometer used and the number of aerosol generators required. The cost of the basic equipment and sufficient smoke-generating capacity for testing a 30,000-cfm system is roughly equivalent to the cost of forty 24- by 24-in. high-efficiency filters in their cartons. (R. W. Schneider)

EMERGENCY DECONTAMINATION PREPARATIONS AT CHALK RIVER

Two previous reviews of decontamination hazards dealt with criticality and radiation hazards during general decontamination,[8,9] and a third was on hazards that had been encountered or which might be encountered during the decontamination of nuclear reactors.[10] The latter review emphasized that very serious consideration must be given at all reactor sites to the possibility that emergency operations may have to be initiated on extremely short notice and that some operations may involve decontamination of persons with high levels of radioactivity. The present review describes the excellent state of preparation for handling nuclear incidents and disasters that has been developed at the Atomic Energy of Canada Limited, Chalk River, Ontario.

At Chalk River, the decontamination experiences resulting from the NRX reactor incident[11] of December 1952 and the NRU incident[9,12,13] of May 1958 made clear the need for improved organization and equipment that would be ready to handle future incidents in a more efficient manner. Such improvements have now been made and are described in a series of papers.[14-16] Not only is the installation now better prepared to deal with large-scale emergency decontamination of personnel, buildings, and equipment, but also the routine decontamination procedures have been organized to ensure a high degree of efficiency.

Facilities

The decontamination building (floor area, 7000 ft^2) described by White[14] is designed for production-line operations. This results in the

safest possible movement of equipment through the decontamination processes. The design criteria that were the basis for planning the building specified the following:

1. Adequate storage space, ventilated and shielded.

2. Separate decontamination rooms for each type of operation, adjacent to the storage areas.

3. Monitoring and shipping areas remote from the storage and decontamination areas.

4. Straight-line flow of items through the building.

5. Clothing change rooms separated from work areas.

6. Office area separate from active areas.

7. Ventilation, well designed and generous in capacity.

8. Modern fume hoods for hand work areas and machinery.

9. Smooth, painted walls, with the minimum of obstructions.

10. Drying rooms for plastic clothing and footwear.

11. Efficient waste disposal facilities.

12. Special room for flammable solvents.

The new decontamination building makes use of a number of machines to lessen handwork and thus reduce personnel exposure. These include a rotary-drum washing machine for clothing, a metal-parts washer that automatically dips up to 750 lb of metal parts into a heated reagent bath, and a 5-kw ultrasonic cleaner with six transducers mounted in a bath 12 × 24 × 40 in. Large and heavily contaminated items are cleaned by steam jets in a closed pit equipped with an offgas system.

Reagents

With regard to reagents, White[14] states that phosphate compounds, as a group, have been found to be the most effective decontaminants for mixed fission products. EDTA (ethylenediaminetetraacetic acid) is also used for plutonium or uranium decontamination. Approximately 90% of all equipment is satisfactorily decontaminated by vacuum cleaning, followed by a quick wash with a warm phosphate solution. More vigorous reagents are used when phosphates are not effective. The phosphate solutions, normally used at 150 to 175°F, include aqueous 2% Calgon with a wetting agent, and they sometimes also contain 2% EDTA. The addition of 0.05% wetting agent (Hydra-Clene) to a 175°F solution of 2 to 5 parts by weight of H_3PO_4 improves the action of the acid. Human skin is decontaminated by soap and water, followed, in resistant cases, by a mixture of 75% Alconox (a commercial mixture of phosphates) and 25% EDTA.

Equipment

Seven special items of equipment recently developed at Chalk River to assist in decontamination and monitoring operations have been described by MacFarlane and Beal.[15]

1. Mobile air exhaust unit. This 1000-cfm unit, mounted on a dolly, draws air through a light, flexible canvas hose that is internally supported by a wire helix. The air drawn from the contaminated location is passed through a high-efficiency filter before it reaches the fan. A second exhaust hose disposes of the filtered air. The unit is useful in many situations in which normal ventilation is inadequate in the presence of airborne contamination.

2. Mobile air sampler. A gasoline-motor-powered air sampler has found many applications where air is sampled in locations remote from electric power. The unit, with a capacity of 10.6 cfm, weighs only 70 lb.

3. Mobile change-room trailer. Among the most important of the special items is the mobile change-room trailer. The unit is a standard, mobile home that has been internally partitioned and equipped to conform to good change-room design practice.

4. Shielded Settling Box and Shielded High-Efficiency Filter. Heavy contamination, with high radiation fields, such as might occur when loose, irradiated reactor fuel escaped from confinement, can often be greatly reduced through vacuum cleaning. A mobile vacuum cleaner with an associated shielded settling box and a high-efficiency filter is available. The settling box and filter may be disposed of without personnel exposure by suspending the units over waste disposal holes and pulling out retaining pins by means of ropes.

5. Steam-Detergent Decontamination Gun. High decontamination efficiencies on many types of surfaces have been obtained with a new type of steam gun. A mixture of steam and detergent is ejected through a slit orifice, only 0.015 in. wide, at 90 psi. The orifice is preferably kept in contact with the surface or at a distance of 1/8 in. and is stroked over the surface in the manner of shaving with a safety razor. Various guns for different applications have been devised. The gun reduces or eliminates the need for hand scrubbing of contaminated walls of buildings, for example, and should fill an outstanding need in the field of decontamination.

6. Mobile, Remotely Controlled, Radiation Monitor. The remotely controlled monitor for high-level radiation fields of the type that would exist after a nuclear accident features a monitoring head mounted on an extensible boom that can traverse in both vertical and horizontal planes. The remote control box is small enough to be held in the hand. Use of this unit, rather than hand monitors, should greatly reduce personnel exposures in locating high-level sources of radiation.

7. Inflatable Tent. An inflatable polyethylene tent, commerically available, has been used for housing bulky objects while they are being decontaminated. The tent is provided with an air supply from a fan. The spread of contamination is thus much reduced or eliminated.

Preplanning

The organization that will be activated at Chalk River in the event of another nuclear accident is described by Neill and Marko.[16] Thought has been given not only to how the cleanup can be effectively managed and directed, but also to the necessary supplies, such as face masks and protective clothing, and how they can be systematically handled and cared for. The authors suggest that every plant handling significant quantities of radioactive material should be prepared for an incident involving the release of radioactive material. For adequate decontamination, they should maintain the following:

1. A list of personnel able to perform certain tasks under conditions of high-level radiation fields and surface and air contamination.

2. An emergency staff to keep records of the personnel used in the clean-up program. These records should include the history of previous

exposures, present exposures, urinalyses, blood checks, etc. If men from outside the project are used, for example, from the armed forces, a medical check should be made on arrival and on leaving.

3. Adequate supplies of respirators and protective clothing and an efficient respirator inspection and clothing decontamination service.

4. A special change room, equipped to handle workers returning from decontamination work, and a skeleton staff trained in its use.

5. Special equipment to handle all types of emergencies, and a staff of men well trained in the use of this equipment. These trained men can be used to instruct others.

6. Facilities for dismantling contaminated equipment, such as valves, electric motors, compressors, etc., in fume hoods with adequate ventilation.

7. A well-equipped central decontamination center where contaminated articles such as respirators, plastic clothing, rubbers, tools, and small items of equipment can be decontaminated using mechanical means of cleaning as much as possible.

Routine Operations

The Canadian investigators have found that, although the specialized decontamination equipment was developed for use after a major plant incident, most of the equipment is now being used in routine decontamination work. Caves, dry-box systems, contaminated rooms, shipping flasks, etc., are all being cleaned with the special equipment, and a considerable saving in man hours of cleaning has resulted.[16]

During 1959, $15.00 worth of equipment was recovered for every $1.00 of direct charge for decontamination. Based on this experience, it is

considered that a well-organized decontamination service should be considered a necessary part of any atomic energy establishment.[14] Not only are large sums of money saved, but health and safety standards are easier to maintain. It would appear that the prominent role of the ordinary scrubbing brush in the decontamination field is coming to an end.

(A. B. Meservey)

FISSION-PRODUCT VAPORIZATION AND DEPOSITION FROM URANIUM CARBIDE

The knowledge of the location and chemical form of the fission products is an obvious requirement for safety of any reactor, but in reactors with unclad fuel elements, this knowledge is even more vital because the fission products continuously leak from the fuel element into the coolant. The stationary reactors in the United States that utilize unclad graphite-base fuel elements with helium as the coolant are the Ultra-High-Temperature Reactor Experiment (UHTREX) and the Peach Bottom Reactor (HTGR). Work is also under way on graphite-base fuel for pebble-bed reactor use. In these reactors the chemical behavior of the fission products in the graphite-base fuel and the helium coolant determines their distribution during normal operation. An understanding of the chemistry in these systems is, therefore, vital to safe reactor design.

Vaporization of Fission Products

An understanding of the influence of chemical reactions in controlling the vaporization of fission products from uranium carbide—graphite fuel requires the determination of which metallic elements are carbide formers and therefore exhibit vapor pressures lower than those of the corresponding metals and whether two fission-product elements react with each other within the fuel elements and thereby lower the vapor pressure of both chemical elements. An example of the reaction of fission-product elements would be cesium and iodine combining to form CsI within the porous graphite lattice of a fuel element. Only the vaporization of fission products will be considered, since the chemical reactions depend on diffusion

behavior, a subject that is beyond the scope of this review. Similarly, the release of fission products through recoil from the fuel elements and erosion of the surface of the fuel material will not be discussed.

Vaporization is a problem in chemical equilibria of the type

$$MC_x(\text{solid}) = M\ (\text{gas}) + xC\ (\text{graphite})\ .$$

Since all chemical equilibria are temperature dependent, all predictions of fission-product vaporization require a knowledge of the temperature profile of the fuel element.

The volatilization behavior and other factors that contribute to the presence of fission products in the coolant are given in Table IV-2. The assumptions used in constructing this table were that (1) within the fuel elements the fission products are either chemical elements or carbides, (2) the fuel element contains some impediment (e.g., a graphite sleeve and a purge-gas stream, as in the HTGR, or the thick pyrolytic graphite coverings on the fuel balls proposed for a pebble-bed reactor) that prevents short-lived nuclides (half life <2 min) formed directly by fission from leaking out into the coolant, and (3) the surface temperature of the fuel elements does not exceed 1500°K (2240°F). The volatility classifications used are defined below:

	Vapor Pressure at 1500°K (atm)
Nonvolatile	$<10^{-6}$
Moderately volatile	10^{-6} to 1
Volatile	>1

The sources of measured thermodynamic properties of the carbides were the compilations of Krikorian[17,18] and of Kelly and King.[19,20] Krikorian's

Table IV-2. The Fission Products and Their Volatility from Uranium Carbide Fuel Elements

Element	Percentage of Total Fission-Product Yield After Irradiation of 10^7 sec (116 days)[a]	Vaporization Behavior and Remarks
Zn, Ga, Ge, As	Combined yield <0.1%	Unimportant; very low fission yield
Se	0.45%	Volatile
Br	0.2%	Volatile
Kr	4.0%	Volatile
Rb	4.0%	Volatile; released to coolant partly through the volatile precursors Kr^{85m} and Kr^{87-91}; much may be retained by chemisorption under certain conditions (see discussion of barium and cesium sorption below)
Sr	12.5%	Moderately volatile; Sr^{88} in coolant through volatile precursors; element is carbide former with an estimated pressure of $10^{-4.4}$ atm at 1000°K and $10^{-1.3}$ atm at 1500°K; however, chemisorption can be significant (see discussion of barium and cesium sorption below) in reducing vapor pressure
Y	5.5%	Whatever gets into coolant comes through precursors, that is, those strontium nuclides that were already out of the fuel elements; otherwise yttrium is not volatile
Zr	32.0%	Nonvolatile; the only radioactive isotopes that may leave the fuel elements are Zr^{93} and Zr^{95} to the extent that the short-lived volatile precursors escaped
Nb	1.7%	Nonvolatile; any fission product of mass number 95 released will be the only source of niobium out of the fuel element

Table IV-2. (Continued)

Element	Percentage of Total Fission-Product Yield After Irradiation of 10^7 sec (116 days)[a]	Vaporization Behavior and Remarks
Mo	18.5%	Nonvolatile
Tc	6.0%	Nonvolatile
Ru	12.5%	Nonvolatile
Rh	1.6%	Nonvolatile
Pd	1.2%	Nonvolatile
Ag, Cd, In	Combined yield ~0.1%	Unimportant because of low fission yield; silver and indium moderately volatile; cadmium volatile
Sn	~0.1%	Moderately volatile; vapor pressure 10^{-5} atm at 1500°K; volatile cadmium and indium precursors would lead to small amount outside fuel elements
Sb	0.05%	Unimportant because of low fission yield; moderately volatile
Te	3.0%	Volatile
I	1.5%	Volatile
Xe	18.7 to 24.2%, depending on thermal-neutron flux	Volatile
Cs	17.6 to 12.1% depending on thermal-neutron flux	Ordinarily volatile, but chemisorption on graphite may be significant in retaining much cesium (see discussion below); volatile precursors from elements of mass numbers 133, 135, 137, 138, 139 will account for most of cesium out of fuel elements

Table IV-2. (Continued)

Element	Percentage of Total Fission-Product Yield After Irradiation of 10^7 sec (116 days)[a]	Vaporization Behavior and Remarks
Ba	6.8%	Volatile precursors account for most of barium outside of fuel elements; element is carbide former, with an estimated pressure of $10^{-5.1}$ atm at 1000°K and $10^{-1.7}$ atm at 1500°K; however, chemisorption on graphite further reduces pressure (see discussion below)
La	6.2%	Nonvolatile, but volatile precursors of mass numbers 139, 140, 141, and 142 will account for some lanthanium out of the fuel element; chemisorption on graphite of Cs^{140} and Ba^{140} may be of great significance in preventing the release of the important gamma emitter La^{140}
Ce	19.0%	Nonvolatile; only radioactivity should come from the Ce^{141} released via volatile precursors; probably no Ce^{143} or Ce^{144} will be released because volatile precursors are too short lived
Pr	4.5%	Nonvolatile; only possible radioactivity associated with praseodymium out of fuel elements would come from the unlikely release of volatile precursors of mass numbers 143 and 144
Nd	17.0%	Nonvolatile
Pm	2.4%	Nonvolatile
Sm	2.2%	Similar in chemical behavior to strontium and barium; moderately volatile; estimated vapor pressure over carbide of $10^{-5.9}$ atm at 1000°K, and $10^{-2.3}$ atm at 1500°K

Table IV-2. (Continued)

Element	Percentage of Total Fission-Product Yield After Irradiation of 10^7 sec (116 days)[a]	Vaporization Behavior and Remarks
Eu	0.22%	Same comments as for samarium; estimated vapor pressure over carbide of $10^{-4.5}$ atm at 1000°K and $10^{-1.6}$ atm at 1500°K
Gd, Tb, Dy	Combined yield <0.1%	Unimportant; very low fission yield; nonvolatile

[a]See ref. 24.

empirical methods[17,18] were used to estimate absolute entropies of the unknown carbides. The standard heats of formation of SmC_2 and EuC_2 were assumed to be the same as the heat of formation estimated for SrC_2.[17] Vapor pressures and thermodynamic data of the elements were obtained from the monograph of Stull and Sinke.[21] Information on fission-product decay chains was obtained from Blomeke[22] and Katcoff.[23] The yield data were taken from Robinson and Krouse.[24]

In summary, the significant fission products that will be present in the helium coolant are the inert gases krypton and xenon, the nonmetals iodine, bromine, tellurium, and selenium, the alkali metals cesium and rubidium, the alkaline earth and quasi-alkaline earth metals strontium, bromine, samarium, and europium, and the nonvolatile metals that are decay products of gaseous precursors yttrium, lanthanum, cerium, and perhaps zirconium.

Chemisorption of Alkali and Alkaline-Earth Fission Products on Graphite

In recent investigations, Zumwalt and his associates[25,26] have shown that cesium and barium can be adsorbed to a remarkable extent on graphite at high temperatures. For instance, at 1000°C, when the vapor pressure of cesium was in the range 10^{-5} to 10^{-6} atm, graphite samples held approximately 12 mg of cesium per gram of graphite. This high affinity of cesium for graphite should be distinguished from the formation of cesium-graphite compounds, which occurs at much lower temperatures.[27]

More thorough adsorption studies[26] have been carried out with barium, and the results are expressed by the equation

$$\log P_{Ba} = \frac{(4.7_6 \log Q - 13.9) \times 10^3}{T} + 4.33 ,$$

where P_{Ba} is the partial pressure of barium in atmospheres, Q is the loading of barium in HLM-85 graphite in milligrams of barium per gram of carbon, and T is temperature in °K. For instance, a concentration of 1 mg of barium per gram of carbon at 1500°K has a barium partial pressure of 10^{-5} atm. The estimated barium pressure over barium carbide at the same temperature is $10^{-1.7}$ atm. Although unmeasured, strontium, samarium, and europium should have similar graphite adsorption behavior to barium, whereas rubidium should be similar to cesium.

Even though these adsorption investigations are at an early stage, the available results indicate that the preponderance of alkali and alkaline earth (including samarium and europium) fission products formed within the fuel elements will be retained on the fuel elements or on graphite jackets (e.g., the graphite sleeves in the HTGR). Furthermore, if sufficient holdup times can be attained for halogen and inert-gas precursors within graphite fuel elements, the hazards associated with Sr^{90}, Y^{91}, Cs^{137}, Ba^{140}, La^{140}, Ce^{141} should be greatly reduced.

Parenthetically, it should be noted that Walton[28] deduced the favorable adsorption behavior and its implications from studies of diffusion of cesium, strontium, and barium in graphite. In the conclusion to his paper, he states "The mobility (in graphite) of certain fisssion products (Cs, Sr, Ba) seems to be associated with high adsorptive properties and it is possible that these very properties will effectively keep a reactor coolant circuit clean."

Deposition of Fission Products

The chemical state of the fission products outside the fuel elements has not been explored in any great detail. In turn, very few useful generalities have been derived that relate the chemical state of fission product with the possible surfaces of deposition. The absence of generalities can be attributed to the large number of variables that influence deposition. A partial set of variables might be temperature, pressure, concentration of fission products, concentration of CO, CO_2, H_2O, H_2 or other coolant contaminants, nature of surfaces (especially on the heat exchangers), size of dust particles, and existence or lack of charcoal beds. The buildup of knowledge in the field of fission-product deposition will probably be rather slow, depending primarily on well-designed in-pile tests. The most complete information will become available after the dismantling of large full-scale reactors or reactor experiments.

It is necessary, however, to rationally predict the disposition of fission products after leakage from fuel elements. Brewer[29] has summarized and predicted the chemical states of fission products in the coolant stream of the HTGR. He predicts: (1) "the non-metals will all be converted to compounds of the alkali and alkaline earth metals," with the halides most likely reacting with cesium to form CsI and CsBr; (2) if the coolant gas is carburizing, "there is the possibility of precipitation of carbides of the four alkaline earth elements as the temperature is reduced"; (3) "Due to the presence of CO in the coolant gas, there is the possibility also of formation of oxides of the alkaline earth and alkali metals"; (4) "upon cooling of the gas one would expect condensation of a mixture of metals, oxides, and carbides along with CsBr and CsI";

(5) much of the condensate will be in the form of a dust that will partially revaporize when carried by the coolant back into the core. Brewer therefore concludes that the inert gases, the alkali metals, and the halides "may act as permanent gases to a considerable extent." Brewer's ideas are almost certainly applicable to any reactor with uranium carbide fuel elements.

Cesium iodide has not been found experimentally, as yet, under simulated reactor conditions; however, von der Decken[30] tentatively proposes CsI to explain the deposition of iodine on graphite held at 1000°C in an irradiation test. The vapor pressure[31] of CsI at this temperature is $10^{-1.07}$ atm. Unless CsI is chemisorbed on graphite (which is doubtful), BaI_2 or SrI_2 would be more probable chemical compounds, since these have lower vapor pressures[32] at 1000°C.

Bromley[33] has very briefly discussed the chemical state of fission products both in the fuel element and in the coolant, but his comments do not augment anything in Brewer's papers.[29]

The out-of-fuel element behavior of yttrium, lanthanum, cerium, and zirconium will be governed by CO and CO_2 impurities in the coolant; that is, these fission products should form oxides. These oxides will probably remain on the walls where the CO and CO_2 reacted with the previously precipitated metals. The vapor pressures of these metals and their corresponding oxides are sufficiently low ($<10^{-11}$ atm) that distillation to the heat exchangers should be negligible.

The deposition properties of unreacted cesium, that is, cesium that has not formed Cs_2O, reacted with anionic fission products or chemisorbed on graphite are very uncertain. Metallic cesium, being relatively volatile,

would be expected to deposit preferentially on the coolest surface, but cesium also has the lowest ionization potential of any element. This means that cesium would deposit randomly and be held in place by donating its valence electron to the conduction band of the underlying metal. As might be expected, responsible scientists differ as to which of the properties will dominate the deposition of cesium. Gluekauf of the DRAGON project believes that cesium will probably deposit on the heat exchanger, oxidize there, deposit new cesium metal layers, oxidize again, and slowly build up a scale.[34] Zumwalt of the HTGR project believes that only a monolayer of cesium, at most, will build up on surfaces.[34]

Conclusions

The dearth of experimental information on how chemical reactions affect vaporization and deposition of fission products can, to some extent, be ameliorated by careful use of basic chemical ideas. Specifically, vaporization behavior of fission products from UC_2-graphite fuel elements can be surmised quite well from basic chemical parameters. On the other hand, there is relatively little information and fewer useful ideas on how chemical reactions affect deposition. The only practical recourse is a program of experimentation involving systematic alteration of variables that should affect deposition. The reasons for such a program have been succinctly stated by Adams and Browning: "The designer of a nuclear reactor must be able to predict reliably the location and form of the entire inventory of radioactive fission products associated with the reactor system if he is to fulfill his obligation to control the hazards produced by these materials."[35] (Stanley Cantor)

PARTICLE AND FISSION-PRODUCT BEHAVIOR IN NUCLEAR ACCIDENTS

The first two sessions of the Third Conference on Nuclear Reactor Chemistry, held at Gatlinburg, Tennessee, October 9–11, 1962 were devoted to problems related to release of fission products in nuclear reactor accidents. A broad spectrum of topics was covered in the 12 papers presented in these sessions, which were opened with remarks by the Chairman, S. A. Szawlewicz, Chief, Research and Development Branch, Division of Reactor Development, USAEC, Washington, on the importance of fission-product release studies in reactor siting and hazards evaluation. He stated that:

> "The release of fission products, as a result of a nuclear reactor accident, constitutes the principal danger in the operation of nuclear reactors. Because of this danger, the nuclear power industry is experiencing considerable difficulty in obtaining government approval for siting large power reactors close to populated load centers. The siting problem is related to the establishment and use, in hazards calculations, of conservative 'upper limit values' for the release of fission products to the atmosphere in conjunction with the description of a maximum credible accident."

He also stated that, in determining the potential hazard to the public resulting from an accidental meltdown, four successive and interrelated problems must be considered. These include the calculation of the fraction of fission products released in the region of the core, the determination of the amounts of released radionuclides that are transported into the building or containment shell, the determination of the time and space behavior of the transported radionuclides, and, finally, the extent of their leakage to the outer atmosphere and their dispersion in the atmosphere. He mentioned that, in preparing hazards analyses, the assumption is made that 100% of the noble gases, 50% of the halogens, and

1% of the solids in the total fission-product inventory of the core at the time of the accident are released into the reactor building or containment structure. Szawlewicz concluded his remarks by saying:

> "Perhaps with continued success in the type of studies which will be described today, we will ultimately succeed in providing more accurate descriptions of fission product behavior during an accident, developing, at the same time, methods for effective prevention of releases into the outer atmosphere. In the event of a release into a containment structure, rapid and reliable fission product control by air cleanup systems must first be proven before credit for reduced danger is granted by safety authorities."

Although the various investigators represented on the program have employed quite diverse approaches to the study of the reactor hazards problems mentioned by the Chairman, the discussions can be conveniently considered under three headings: particle production and behavior, fission-product release and behavior, and containment and removal of fission products from the atmosphere.

Particle Production and Behavior

Studies of the effect of temperature and air velocity on the nature of oxide particles formed from small, unclad uranium cylinders oxidized in air at furnace temperatures of 400, 600, 800, 1000, and 1200°C were reported by Coleman and Schwendiman[36] in the only paper devoted entirely to particle production. The principal findings of these investigations are summarized below:

1. The physical nature of the uranium oxide formed changed from fine, black, platelet-like particles (400–600°C) to coarse, coke-like granules (600–800°C), and finally to a dense, columnar, needle-like structure that remained largely intact at 1000 to 1200°C. There was a general increase in oxide particle size with increasing temperature.

2. Less than 4% of the oxide formed at 400 to 1200°C was in the less than 10-μ particle-size range in any of the cases investigated. Mass median diameters of approximately 100 and 160 μ were determined for oxide residues at 400 and 600°C. At 800°C and above, 75% or more of the mass was associated with particles greater than 250 μ in diameter.

3. At air velocities of 8.3 cm/sec and lower, the quantity of particulate oxide reaching a point 20 in. downstream was greatest at 1200°C. The entrained oxide in this case consisted primarily of agglomerates of individual particles from 0.015 to 0.5 μ in diameter that were believed to arise from sublimation of UO_3. As air velocities were increased, aerodynamic entrainment of somewhat larger particles over-shadowed release by direct volatilization.

4. At temperatures of 400, 600, and 800°C, the average diameter of oxide particles reaching a point 20 in. downstream was on the order of 2 to 5 μ at an air velocity of 8.3 cm/sec.

5. Specimens oxidized at 1000 to 1200°C gave rise to few micron-size oxide particles during heating. On rapid cooling, stresses within the system caused projection of microscopic and macroscopic particles at high velocities.

Chamberlain[37] described the results of experimental wind tunnel work in which deposition of particles of various sizes was measured on surfaces of varying degrees of roughness, including aluminum and grass. His data correlated well with the theory of Owen[38] for particles having a size greater than 0.1 μ. This theory treats the transport of a particle through the boundary layer by the impetus given it by turbulence. This mechanism is considered to predominate over transport by Brownian

diffusion. For particles of size smaller than 0.1 μ, deposition is greater than that predicted by Owen's theory, with the result that the deposition velocity is nearly independent of particle size over the range 10^{-3} to 10^{-1} μ. Deposition velocities are greater for molecular iodine and for particles larger than 0.1 μ.

Work on millimicron-size aerosols of the kind observed in in-pile fuel-melting experiments was described by Browning and Ackley.[39] A method was developed for determining the particle-size distribution of radioactive aerosols by diffusion-coefficient measurements, making use of the deposition distribution of the radioactive aerosol on the walls of a channel through which there is laminar flow of a gas. The method is capable of discriminating between various sizes of particulate material over the range 0.004 to 0.3 μ and radioactive vapors. Radioiodine vapor and particles labeled with radioactive iodine were used in these tests.

An apparatus for investigating interactions between radionuclides and aerosols and methods for measuring aerosol particles were discussed by Craig.[40] Operational tests of the apparatus were described in which a radioactive iron oxide aerosol was used to test the stability and reproducibility of the system. Plans for investigating the interactions between various fission product nuclides and several aerosols under a variety of conditions were discussed.

Fission-Product Release and Postrelease Behavior

Because of the major role of clad UO_2 fuel elements in the current power reactor program, most of the emphasis in the fission-product hazards

studies described by Parker and his co-workers[41] has been on appraisal of the effect of various factors on the three processes for the release of fission products by UO_2: diffusion, oxidation, and melting. These factors include the quantity of fuel involved and the effects of burnup, plateout, chemical state, etc.

Since each of these parameters may exert a significant effect on the others, a series of investigations covering a wide range of conditions is being conducted. Significant results were reported for studies of the effect of burnup on three processes: (1) diffusion at high temperature in pure helium; (2) oxidation in air below 1200°C; and (3) melting in air, helium, and CO_2. The burnup effect was found to be most pronounced in the diffusion process and to lead to increased releases in all cases, ranging in value up to factors of 10. Factors of 2 increase were typical in the oxidation process, and only fractional increases occurred during melting.

The only example of a significant effect of cladding on the release of fission products from UO_2 observed thus far was found on comparing release values obtained on melting Zircaloy-clad and stainless steel-clad UO_2 specimens. The Zircaloy-clad specimens released less tellurium by a factor of 30 and less iodine and cesium, both by a factor of 2. Ruthenium release was also reduced by a factor of up to 100 in both metal-clad specimens when compared with release from a miniature sample melted by the carbon-arc image method with traces of air present. It was concluded that the volatility of ruthenium was probably a function of its rate of oxidation to RuO_4.

Small variations were found to occur in the size and shape of uranium oxide particles formed by the vaporization of UO_2 during melting in helium, air, and CO_2. Much more significant, perhaps, was the difference in quantity vaporized as a function of the quantity of UO_2 melted. This varied from as much as 20% from miniature samples (30 to 50 mg) to as little as 0.2% from 30-g melts.

Roberts and his co-workers[42] presented the results of a series of in-pile experiments in which miniature fuel elements were destroyed by melting or combustion, and the release of fission products was determined. These studies were conducted in order to compare fission-product release behavior in an operating reactor with that observed in out-of-pile studies. Miniature stainless steel-clad UO_2 fuel elements were melted by heat of fission in the Oak Ridge Research Reactor in a helium atmosphere. Nearly all the iodine, tellurium, cesium, krypton, and xenon were released from the fuel, as well as more than half of each of the other fission products. Much lower fractions of fission products were released from the high-temperature zone of the furnace, with the strontium, zirconium, ruthenium, barium, cerium, and uranium being almost completely retained. Those parts of the fission products that were deposited in the exit gas line produced distributions corresponding to particles 22 and 30 A in diameter. In one experiment in which a miniature uranium carbide-graphite fuel specimen was burned in air in the reactor, only a small fraction of the strontium, zirconium, barium, cerium, and uranium was released from the hot zone of the furnace, but large amounts of ruthenium, iodine, tellurium, and cesium were released. It was concluded that, in a reactor accident, although fission products will be released copiously from the

fuel material when it is destroyed by melting or burning, they may be expected to be retained almost entirely inside the high-temperature zone, with the exception of noble gases, iodine, cesium, and tellurium, and, if oxygen is present, ruthenium.

Efforts to employ a plasma jet for fission-product release rate experiments with $BeO-UO_2-Y_2O_3$ fuel materials were discussed by Conn and his co-workers.[43] It was found that the samples could not be held in a molten puddle, and the high velocity of the plasma jet caused uncontrollable erosion of the molten portion of the specimens. These difficulties, in addition to deposition of a large fraction of the activities in the apparatus prior to the charcoal trap, made release-rate experiments impractical by this method. It was found that iodine release was extremely rapid when the specimens were melted.

Two other reports on in-pile tests of tubular $BeO-UO_2$ fuel specimens stabilized with Y_2O_3 were also presented.[44,45] Data were discussed which demonstrated that recoil release of fission products is the dominant release mechanism in dry air at temperatures up to 1400°C. In this paper,[45] a comparison of fission-product release rates in wet and dry air was presented. Iodine, barium, and strontium were the principal elements studied. It was reported that the hydrolysis of BeO in these fuel materials at temperatures of 1375°C and below results in loss of fission products and redeposition of BeO in cooler regions of the fueled tubes or test assembly, but it was pointed out that the hydrolysis can be effectively prevented by the use of ceramic coatings.

Castleman[46] reported data on fission-product deposition as a function of wall temperature. Specimens of U–3.5% Mo alloy, irradiated at

tracer level, were melted at 1200°C in air or in helium purified by passage through a bed of titanium chips at 850°C, and the quartz tube was sectioned for radiochemical analysis of the fission products deposited on the wall. The most interesting behavior was exhibited by iodine. In purified helium, less than 10% of the released iodine deposited in regions below 200°C, and most of it was found where the wall temperature was in the range 250 to 350°C. In the presence of air, 90% of the released iodine deposited in regions at temperatures below 50°C and the major portion of it collected in the cold charcoal trap. On the basis of experimental observations and thermodynamic estimates, it was concluded that the probable chemical state of iodine above molten uranium in an inert atmosphere is UI_2 and I; in air, elemental iodine is formed. In a helium atmosphere, barium-lanthanum and cerium deposited in regions at temperatures ranging from 1000 to 1200°C, but in air very little of these species was released, presumably because of the formation of nonvolatile oxides. Molybdenum, on the other hand, volatilized in air because of the formation of MoO_3, but none was released in helium. No adequate explanation was found for the behavior of tellurium. In helium, tellurium was found throughout the tube, with the peak activity in the zone that was at 600 to 850°C. Elemental tellurium, under the same conditions, deposited in a wide band of the tube with temperatures that ranged from about 225 to 525°C. The behavior of this element in air was not reported.

The role of meteorology in coping with the aftereffects of a nuclear accident was described by Islitzer.[47] The nuclear accident at the SL-1 reactor on January 3, 1961 at NRTS occurred under meteorological conditions

that favored maximum transport of any airborne effluent with minimum dilution. Unfortunately, these adverse meteorological conditions existed in varying degrees for several weeks following the accident while I^{131} was being released from the reactor at a slowly decreasing rate. Continuous surveillance and analysis of weather conditions were instituted shortly after the accident and were continued until the need no longer existed. Information obtained from the studies, together with results of aerial and ground radioactivity surveys, permitted estimates of the I^{131} release rate to be made, correlations of air concentrations and surface deposition to be studied, and the relation between observed radiation patterns and meteorological information to be described. Although only an estimated 80 curies of I^{131} was released during the accident and the three weeks following, iodine was detected throughout the entire environmental monitoring network extending 100 miles down the Snake River Plain from the SL-1. Comparison of predicted and measured average air concentrations of I^{131} for this period were in reasonable agreement up to several miles, but predicted values were a factor of ten or more lower than the measured values at distances beyond 15 miles. It is believed that the confining effects of the nearby mountains and the diurnal variations of winds in the region prevented the proper use of normal atmospheric diffusion equations for this purpose. A complete analysis of the meteorological aspects of this accident has been reported.[48]

Fission-Product Containment and Removal from the Atmosphere

Experimental studies intended to provide economic safety devices for reducing the consequences of radioactive releases were described by

Silverman.[49] One concept, termed "foam encapsulation," involves the use of an aqueous, plastic, or rigid foam to encapsulate halogens and particulate matter evolved within a reactor containment vessel or reactor building. Only aqueous foams (1000:1 lauryl sulfate solution containing sodium thiosulfate) have been studied to date. Results obtained with iodine concentrations of 13 ppm released inside a 1200-ft^3 chamber indicated that as much as 90 to 95% of the iodine was removed from the atmosphere within 10 min. Removal of 0.2 μ and 0.074 μ uranine aerosols appeared equally effective. Dynamic studies of foam continuously injected into a duct containing an aerosol indicated removal of an average of 80% of the particles within a duct length equivalent to 5 to 10 pipe diameters. Radiation stability of the expanded foam is currently being studied, and large-scale tests involving I^{131} in curie amounts will be conducted at the Oak Ridge National Laboratory in the near future.

The second concept discussed by Silverman concerned use of a "diffusion board" for reactor containment. This cell-like structure would replace the metal plate in a containment vessel and would be pervious to expanding gases but impervious to particulate matter and iodine vapor. Other necessary properties include noncombustibility and resistance to steam, shock waves, and nuclear radiation. Tests and evaluations of several materials having some of these properties are under way.

The diffusion-coefficient measurement technique described by Browning and Ackley[39] was applied in studies of the efficiences of various gas-cleaning methods for removal of millimicron-size radioactive aerosols. It was found that radioactive particles in this size range were capable of penetrating conventional particulate filters and adsorbers. Fog

nucleation and aqueous foams similar to those discussed by Silverman were tested as removal methods and found to be promising.

Beattie[50] reviewed both the nature of the problem of containment of gaseous and particulate fission-product discharge from large power reactors in the United Kingdom and the economic need to advance the program in the interest of more favorable reactor siting, as well as its extension to newer reactor types. In the past the British research and development program on removal of fission products has been concerned entirely with the Magnox-type fuel system (magnesium-clad uranium) cooled with pressurized carbon dioxide. From this system the ingestion hazards are limited to partial releases of volatile iodine, tellurium, and cesium. With increasing reactor size, however, the amount of "gamma shine" contributed by the rare gases trapped in the containment system becomes almost the limiting factor. A 1000-Mw reactor will contain up to 100 million curies of rare gases and more than 1 kg of chemical iodine. Such quantities of radioisotopes cannot be adsorbed on the available surfaces of the containment shell or allowed to remain unshielded in the building, since they would constitute a formidable gamma source capable of delivering a large dose at a distance of 1/2 mile or more. It would be desirable then to advance the confinement process to a point where both iodine and the rare gases could be reduced to a small volume readily adapted to shielding.

While some problems still exist that challenge the effective use of charcoal beds and filters for reactor gas treatment, such as overheating and the resultant migration of iodine in charcoal beds and plugging of filters with large quantities of airborne particles, the dry system for

air cleaning appears to be a reasonable solution. Water (cooled or moderated) reactors, however, threaten to complicate these systems with the possibility of water logging by steam. For such reactors, it would perhaps be profitable to consider a reinvestigation of aqueous scrubbers, which appear to be promising for the simultaneous removal of condensate and large quantities of released iodine, tellurium, and cesium. The fission products, except for the rare gases, could thus be concentrated in the small volume so urgently needed for shielding purposes. (R. E. Adams, C. J. Barton, W. E. Browning, Jr., and G. W. Parker)

CONTAINMENT OF GAS-COOLED POWER REACTORS

H. N. Culver*

All nuclear reactors have similar basic provisions for containment of activity. These basic provisions take the form of barriers that prevent or minimize activity release and special equipment that collects and relocates any activity that is confined in zones between the barriers. The barriers that prevent or minimize activity release are the fuel material, the fuel element cladding, the reactor coolant system, and finally the containment building, which is designed to release activity at a rate that will not exceed an acceptable level, as defined by conditions unique to the location of the particular reactor. All power reactors have these basic barriers for retaining activity, but their relative importance varies with the type of reactor.

The gas-cooled power reactors that have been proposed or constructed to date have certain basic characteristics that differ appreciably from those of other reactor types. In general, the gas-cooled reactor has a large core that contains mostly ceramic material. The result is that the gas-cooled reactor has a low power density and a high heat capacity.

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This may be seen in Table IV-3, in which several gas-cooled reactors are compared with a typical water-cooled reactor. As indicated, the power density for the Yankee plant is 5 to 100 times the power density of a number of gas-cooled reactors, but, for a given power level, the heat capacity is appreciably greater for gas-cooled reactors. The power density and the heat capacity of the core greatly influence the integrity of the fuel and cladding at the time of an accident, and therefore the containment provisions are intimately tied to these factors.

For most power reactors, the worst accident would be depressurization accompanied by loss of the coolant through a breach in the system piping. The initial period when the coolant was being expelled from the system would be of great importance with respect to the adequacy of containment, for it would be during that period that the fuel elements would be subjected to the most severe temperature transients. During the depressurization period and for short periods thereafter, it would be difficult to be assured that adequate cooling was maintained in the core region; and, therefore, in analyses of accidents, it is common to assume, conservatively, that no flow exists for some finite period. Under this assumed condition, the fuel element cladding would be subjected to a temperature transient that would be quite rapid and could, in some cases, cause fuel element failures. Following the initial rapid rise, the cladding temperature would decrease slowly as the heat radiated to the core, and it ultimately would follow the core temperature. There would be a slow temperature rise in the core from the decay heat in the fuel. As may be seen in the comparisons in Table IV-3, the initial average rate of temperature change in the core for various reactors

Table IV-3. POWER-DENSITY AND HEAT-CAPACITY CHARACTERISTICS OF SOME POWER REACTORS

Reactor	Thermal Power (Mw)	Average Power Density (w/cm^3)	Heat Capacity in Core (cal/°C)	Core Temperature Rise	
				Initial (°F/min)	At 1 hr (°F/min)
Yankee[a]	392	59.3	2.48×10^6	203	71
Berkeley[b]	555	0.56	4×10^8	1.8	0.6
EGCR[b]	84.3	1.88	2.37×10^7	4.6	1.6
Peach Bottom[b]	112.5	8.15	7.96×10^6	18.2	7.4
Julich[b]	46	2.16	8.20×10^6	7.3	2.6
Dragon[b]	20	13.9	8.62×10^5	29.9	10.5

[a]Pressurized-water reactor.

[b]Gas-cooled reactors.

approximately 30 sec after an accident would be extremely small for reactors with low-heat capacity. Table IV-3 also shows that the rate of temperature change would decrease by a factor of 3 after a period of 1 hr. Thus it may be seen that a major difference exists between gas-cooled and water-cooled reactors with respect to the temperature conditions predicted for the core following a depressurization accident. As a result of the relatively slow temperature rise that would occur in the core of a gas-cooled reactor, the containment system design is usually based on the assumption that the integrity of most of the fuel elements will be maintained and that the most effective barrier for retaining activity will be the fuel rather than the containment structure. In order to assure the long-time integrity of the fuel, however, it is necessary to provide some form of cooling for long periods of time. This is done in several ways, as indicated below in discussions of the specific reactors.

Although the use of graphite for the moderator is of great benefit in limiting the temperature rise in the fuel immediately following an accident, the graphite introduces other problems that are somewhat unique to the gas-cooled reactor and which significantly influence the integrity of various containment barriers. When graphite is at elevated temperatures, its reaction with oxygen can have serious consequences. Graphite oxidation can lead to higher core temperatures which, in turn, can significantly affect fuel element integrity or, in those cases where fuel and moderator are combined, the oxidation of the graphite may lead directly to the release of activity. In addition, the products of combustion can add to the pressure in the containment structure. Therefore, for

containment integrity, not only fuel integrity but adequate control over combustion in the core must be considered.

Containment Provisions for Specific Gas-Cooled Reactors

Information pertaining to the containment provisions of specific gas-cooled reactors is given in Table IV-4, in which the reactors listed are divided into five classes. Each class has different design conditions that significantly alter the fission-product retention of the fuel material and the fuel element cladding. Therefore each class places different dependence on the retention and location of activity, both during normal operation and during credible accident conditions. The design data presented in Table IV-4 were obtained from a number of sources,[51-63] and, although the design values indicated for a specific reactor may differ slightly in the final design or following operation, the numbers shown are close enough to the actual values to be useful in comparing the containment provisions.

Class A: Metallic Fuel with Metal Cladding. The largest group of gas-cooled reactors include those that utilize metallic fuel with metal cladding. The fuel material is either metallic uranium or a uranium-molybdenum alloy, and the cladding material is Magnox or a magnesium alloy. The maximum fuel temperatures are in the range 1000 to 1100°F, and the maximum cladding temperature is generally in the range 750 to 850°F. (The Latina and the Japan GCR operate with cladding temperatures up to 910°F.) As may be seen in Table IV-4, all these reactors are cooled with CO_2 and are graphite moderated.

Fission products are effectively retained in the fuel material during normal operation. If a defect develops in the cladding, a small

Table IV-4. COMPARISON OF DESIGN DATA PERTAINING TO CONTAINMENT OF GRAPHITE-MODERATED GAS-COOLED REACTORS

Reactor Class	Reactor Plant	Thermal Power Level (Mw)	Fuel Material	Maximum Temperature of Fuel Material (°F)	Fuel Element Cladding Material	Maximum Temperature of Cladding Material (°F)	Reactor Coolant	Reactor Coolant System Pressure (psia)	Reactor Outlet Temperature (°F)	Approximate Fission-Product Activity in Coolant During Normal Operation (curies)	Assumed Activity Release From Fuel During MCA[a] (curies)	Containment				References
												Type of Containment	Leakage Rate (%/day)	Design Pressure (psig)	Design Temperature (°F)	
A	Calder Hall	220	U metal	1040	Magnox-Al2	815	CO_2	95.4	637	0	(b)	Conventional building				51, 62
	Chapelcross	220	U metal	1040	Magnox-Al2	815	CO_2	95.4	637	0	(b)	Conventional building				51, 62
	Berkeley	555	U metal	1022	Mg alloy	806	CO_2	140	653	0	(b)	Conventional building				51, 59
	Bradwell	531	U metal	1017	Mg alloy	817	CO_2	142	734	0	(b)	Conventional building				51, 62
	Hunterston	569	U metal	1076	Magnox	849	CO_2	160	764	0	(b)	Conventional building				51, 62
	Hinkley Point	960	U metal	1040	Magnox-Al2	806	CO_2	195	705	0	(b)	Conventional building				51, 62
	Trawsfynydd	870	U metal	1085	Magnox-Al2	824	CO_2	247	750	0	(b)	Conventional building				51, 63
	Latina	705	U metal	1085	Magnox-Al2	904	CO_2	192	734	0	(b)	Conventional building				51
	Japan GCR	585	U metal	1135	Magnox	910	CO_2	208	739	0	(b)	Conventional building				51
	Dungeness	835	U metal	1135	Mg alloy	828	CO_2	283	770	0	(b)	Conventional building				51
	Sizewell	947.7	U metal	1068	Magnox-Al2	849	CO_2	282	770	0	(b)	Conventional building				51
	G2	200	U metal	1035	Mg-Zr alloy	780	CO_2	213	535	0	(b)	Conventional building				51, 62
	G3	200	U metal	1035	Mg-Zr alloy	780	CO_2	213	572	0	(b)	Conventional building				51, 62
	EDF-1	300	U-Mo alloy	1022	Mg alloy	750	CO_2	377	653	0	(b)	Pressure vessel	(b)	9.5	(b)	51, 60, 62
	EDF-2	792	U-Mo alloy	1094	Mg alloy	775	CO_2	365	688	0	(b)	Conventional building				51, 62
	EDF-3	1193	U-Mo alloy	1100	Mg alloy	815	CO_2	305	730	0	(b)	Conventional building				51
B	EGCR	84.3	UO_2	3000	Stainless steel	1450	Helium	313.7	1043	0–1000	85,000	Pressure relief	0.3	9	200	58
	AGR	100	UO_2	3100	Stainless steel	1382	CO_2	282.5	1050	(b)	(b)	Pressure vessel	0.1	10	(b)	51, 54

Table IV-4 (continued)

Reactor Class	Reactor Plant	Thermal Power Level (Mw)	Fuel Material	Maximum Temperature of Fuel Material (°F)	Fuel Element Cladding Material	Maximum Temperature of Cladding Material (°F)	Reactor Coolant	Reactor Coolant System Pressure (psia)	Reactor Outlet Temperature (°F)	Approximate Fission-Product Activity in Coolant During Normal Operation (curies)	Assumed Activity Release From Fuel During MCA[a] (curies)	Containment				References
												Type of Containment	Leakage Rate (%/day)	Design Pressure (psig)	Design Temperature (°F)	
C	HTGR	112.5	UC_2-ThC_2-C	2730	Graphite	2160	Helium	350	1354	109–4225	5.2×10^6	Pressure vessel	0.2	8	150	57
	Dragon	20	UC_2-ThC_2-C	2880	Graphite	1850	Helium	294	1382	270	(b)	Pressure vessel Inner shell Outer shell	 0.1 5.0[c]	 10 3	 100 95	51, 55, 56
D	Julich	46	UC in graphite	2696	Graphite	1958	Helium	150	1562	4.4×10^5	4.8×10^5	Pressure vessel	2.5	29.4	212	51, 52, 53
E	UHTREX	3	UC or UC_2 in graphite	3200	None		Helium	500	2400	(b)	$\sim 8 \times 10^6$	Pressure vessel	1.0	7.5	(b)	61

[a]The values shown in this column were obtained from the sources referenced. Since the assumptions vary greatly in the referenced documents, comparison of the numbers alone is not meaningful. The reader should refer to the reference document for more complete information.

[b]Information not available in the references cited.

[c]Value given in %/day per psi of pressure differential.

amount of activity may be released, either by recoil or as a result of the coolant reacting with the fuel material. In general, the reactors in this class do not have a significant amount of fission-product activity in the coolant during normal operation. They operate with gas outlet temperatures in the range 650 to 750°F, and therefore the graphite temperatures in the core are significantly below those required for runaway graphite oxidation.

Fuel element cladding temperature transients during accidents are limited in these reactors so that in the event of an accident there would either be no fuel element failures or only a small number. The temperature rise in the cladding for the worst accident conditons would depend on the location of the fuel element, the heat radiation coefficient of the cladding, and the time required to scram the reactor.[63] Temperature transients in this class of reactor are limited to about 1160°F in order to prevent combustion of the cladding.

All the Class A reactors, with the exception of the EDF-1, utilize conventional structures as the containment building. (The EDF-1 reactor was built with a pressure-vessel as the container for experimental reasons). Based on the limited amount of available information pertaining to accident analyses for these reactors, it may be concluded that the fuel material and the cladding constitute the main barriers for retaining fission products. These reactors, when operating in the temperature ranges indicated in Table IV-4, do not require additional barriers for retaining activity, other than the conventional building, if adequate provisions have been made for a rapid scram and sufficient heat removal during the period of the accident to prevent subsequent

fuel element failures. Such emergency cooling is provided either by natural convection of CO_2 or by forced convection of air.

Class B: Ceramic Fuel with Metal Cladding. In attempts to obtain higher temperature gas in gas-cooled reactors, the metallic fuels (Class A) were replaced with ceramic fuels and the Magnox cladding was replaced with stainless steel (Class B). The ceramic fuels, such as UO_2, can operate at considerably higher central temperatures because of their low thermal conductivity compared with that of uranium metal and because of the poor heat transfer across a small gap between the fuel and the cladding. In the EGCR, fission products will be released from the UO_2 by diffusion at UO_2 temperatures up to 2912°F. At higher UO_2 temperatures, it is postulated that activity may also be released by metallurgical changes, such as grain growth.

The design data for the EGCR and the AGR are similar in that both reactors utilize UO_2 fuel and stainless steel cladding. The fuel temperatures are similar, as are the coolant outlet temperatures. Both the AGR and the EGCR have fuel elements that effectively retain most of the fission products within the fuel material during normal operation. If there were a defect or failure in the cladding, only a small amount of activity would be released. It is estimated that in the EGCR a maximum of 100 curies could be released from the fuel and enter the coolant following a fuel cladding failure during normal operation. Since helium is to be used as the coolant in the EGCR, there will be no chemical reaction between the fuel material and the coolant following fuel element failures during normal operation. Although no information is available concerning the activity expected in the coolant in the AGR, the mechanisms

for release of activity to the coolant will be the same as for the EGCR, with one exception. In the AGR, the coolant could react at a slow rate with the UO_2 following the failure of the cladding. Therefore the amount of activity buildup in the coolant in the AGR will depend on the residence time of the failed element in the core.

Although the fuel and cladding will be effective barriers for retaining fission products during normal operation, a small number of the ceramic-fueled metal-clad fuel elements may fail under specific accident conditions that may be postulated. In fact, the fuel elements in Class B reactors would be more likely to fail at the time of a maximum credible accident than the fuel elements in Class A reactors, because the fuel material will operate at temperatures significantly above the maximum cladding temperature. As may be seen in Table IV-4, in the case of the Class A reactors, the difference between the maximum cladding temperature and the maximum fuel temperature is less than 300°F; however, in the case of the Class B reactors, the temperature difference is of the order of 1550 to 1700°F. If it is postulated that at the time of the mca there will be no coolant flow in the core for a short period of time, the cladding temperature will rise very rapidly. The rate of rise will depend on such factors as the time required to effect a scram and the emissivity of the cladding, but it will be greatly influenced by the temperature difference between the cladding and the fuel at the time coolant flow is disturbed. In addition to the increase in the cladding temperature, the failure of the cladding will be influenced by the stresses in the cladding during the transients. The cladding in Class B reactors will fail by stress-rupture, which is dependent upon both the

stress level and the temperature. The stress level will be determined by the pressure buildup inside the fuel element, as well as the geometry of the fuel cladding. In the EGCR, it is conservatively estimated that as many as 330 fuel elements could fail at the time of the mca. This would result in an initial release of an estimated 26,000 curies of activity. Following the failure of the fuel elements, additional activity might be released over a period of time as the UO_2 in the failed elements oxidized by contact with air that would enter the core. It is conservatively estimated that the release by oxidation would result in a maximum activity external to the coolant system of approximately 85,000 curies. This amount of activity would be in the containment building at 96 hr after the accident began. No information is available regarding the postulated number of fuel failures in the AGR or the resultant activity release that could occur for the postulated mca for the reactor; however, the similarity between the EGCR and the AGR suggests that the maximum credible accident for the AGR could lead to some fuel element failures and the associated activity release from the fuel.

The containment provisions for the AGR and the EGCR are similar in many ways. The fuel material, cladding, and primary system barriers are similar, and both reactors utilize a containment shell that is designed to withstand the pressures that would result from the postulated accidents. Some of the differences in the containment provisions for the two reactors external to the primary system are of interest and are discussed below.

In the AGR, the entire reactor coolant system will be located inside a concrete structure called the inner containment vessel. This inner containment vessel will serve the function of controlling the location of activity released from the primary system and also act as the biological shield. During normal operation, air will enter the inner containment vessel and sweep the reactor vault and heat exchanger enclosures. A small part of this flow will be discharged through filters and up the stack; the remainder will be recirculated. Gaseous discharges that could result from a depressurization accident may be vented to a chemical-absorption plant, which will contain a heat sink, filter, and activated carbon packs. The heat sink is designed to cool the contents of the CO_2 system plus the contents of one steam generator to about 176°F before passage into the filter system. Valves will prevent discharge up the stack. Either a pressure rise in the containment vessel or high gamma activity will automatically connect the discharge from the inner containment vessel to the chemical plant circuit. A 5-psi rise or a release rate exceeding 100 curies of gamma activity for 30 min will close the vent to the stack and open the system for recycle operation. The inner containment vessel will be protected with relief valves set at 5 psig. The outer containment vessel, or the containment building, is designed to withstand a pressure of 10 psig and is specified for a leakage rate of 0.1%/day. With the above provisions, much of the activity will be removed by the chemical plant before the gases containing activity are discharged into the outer containment building. Much of the iodine activity will also be retained in the inner containment vessel because of deposition. The discharge of the contents of the CO_2 system,

as well as one heat exchanger, to the heat sink prior to release to the inner containment vessel should result in an appreciable reduction in the pressure inside the containment building, as compared with direct discharge to the outer containment vessel. This method of operating the containment system is illustrative of the combination of pressure containment and the pressure-suppression containment.

In the EGCR, the entire reactor coolant system is located inside a concrete structure that has the primary purpose of providing adequate shielding for the coolant system. As in the case of the AGR, the compartments are ventilated with air. Unlike the AGR, however, in the EGCR, in the event of the depressurization of the primary system, gases will be released directly to the containment building. During normal operation, air will be swept through the various compartments and discharged to the stack. Shield-cooling air will be passed through an absolute filter and an iodine removal unit prior to discharge to the stack. In the event of high activity in the stack or low pressure in the coolant system, valves will close to seal the containment building. Following isolation of the containment building, fans will recirculate the contents of the containment building, and the filters and iodine-removal unit in the recirculation system will reduce the activity concentration inside the building. The EGCR design provides for an emergency cooling loop, which will be placed into operation in the highly unlikely event of complete failure of all main blowers. When this loop is placed into operation, a nitrogen purge will be supplied to the core to minimize oxidation of the graphite. As this purge is supplied, the pressure inside the containment building will increase. Provision was therefore made in the design to discharge

the contents of the containment building at a controlled rate through a fission-product removal system and then to the stack. This venting will not begin until 24 hr after the initiation of the accident. The design also provides a water spray on the containment building to assist in the reduction of pressure and temperature.

In both the EGCR and the AGR, provisions are required for the removal of afterheat in order to ensure the long-term integrity of the core and the fuel elements. In the AGR a CO_2 blanket will be introduced into the core, and in the EGCR nitrogen will be used to purge the core. In both cases the purge gas will be added to prevent oxidation of the graphite. In the case of the AGR, with only 10% air in the core, natural convection will be adequate to cool the core. In the EGCR some forced circulation will be needed for a period of up to 90 days, although the purging will be required only for a short period of time.

Class C: Ceramic Fuel with Ceramic Cladding. In order to achieve even higher temperatures, reactors have been designed that utilize all-ceramic fuel elements. As indicated in Table IV-4, the Peach Bottom reactor (HTGR), the Dragon reactor, and the Julich reactor are of this type. Because of the very basic differences between the Julich reactor and the HTGR and Dragon, the Julich reactor is discussed separately as a Class D reactor.

It may be seen in Table IV-4 that the design data for the Dragon reactor and the HTGR are similar and that both reactors will have appreciably higher reactor outlet coolant temperatures and cladding temperatures than those for the EGCR or the AGR. The lower fuel material temperature results from the improved thermal conductivity of the carbide fuels as

compared with the oxide fuels; however, both the Dragon and the HTGR will have higher average core temperatures than the AGR or the EGCR. The fuel elements for both the HTGR and the Dragon consist of uranium carbide and thorium carbide in a graphite matrix. Both reactors will have purge systems that will provide a continuous sweep of gas through the fuel elements to transfer fission-product gases to internal and external fission-product traps. Two types of fuel particles are proposed for this class of reactor: uncoated and coated. The analysis of hazards for the HTGR was based on the use of uncoated particles, although the reactor fuel will probably contain coated particles. It may be seen in Table IV-4, that the Class C reactors differ from the Class A and Class B reactors in that no attempt will be made to retain all the fission products in the fuel during normal operation. Rather, the fuel elements are designed to retain most of the nonvolatile fission products but to allow significant quantities of the noble gases and some volatile fission products to escape from the fuel material. Activity released from the fuel will be relocated in traps, and only a small amount of activity will be circulated in the coolant system. In the Dragon reactor, at equilibrium, about 96% of the gaseous fission products will be located in the fission-product traps and only about 4% of the gaseous activity will be in the fuel.[55] About 270 curies of activity will be in the coolant system. In the HTGR most of the activity generated in the fuel will remain in the fuel; with uncoated particles about 10% of the activity will be released, whereas with coated particles less than 2% of the activity will be released. Of the activity released, 75 to 80% will be held up in the internal traps, with the remainder being in the external trapping system. The activity level in the coolant of

the HTGR is estimated as 109 to 4225 curies. The lower value of the activity is for the case of all fuel elements operating in the normal condition and is based on fission-product release data for coated fuel particle compacts after burnups equivalent to 50% of the fuel element life. The higher value is approximately 20 times the calculated activity based on uncoated fuel particle compacts and is the basis for the primary system shielding design. The maximum allowable activity in the HTGR coolant averaged over a period of one year is specified as 1800 curies.

Thus both the Dragon and the HTGR will operate with some activity in the coolant at all times. Although the EGCR and the AGR will have activity in the coolant during some periods of operation, the level of activity will, in general, be made lower than for the Dragon or the HTGR.

The estimated activity release from Class C reactors at the time of an accident, as in the case of the Class B reactors, determines the requirements of the final containment barrier. Little information is available pertaining to the possible release of activity from the Dragon reactor in the event of an accident, but the hazards analysis for the HTGR indicates that the containment building for the HTGR is adequate for a release of about 5.2×10^6 curies. In order to obtain this activity release, however, a number of unlikely events must occur.

The amount of activity released from the HTGR in the event of an accident will depend upon different factors than in the case of the Class B reactors. The use of the ceramic cladding will effectively eliminate the problem of cladding failures; however, in the HTGR, as well as in the Dragon, an accident could result in the loss of the purge flow in the fuel elements and consequently increased coolant activity. If the accident

resulted from coolant system depressurization, which, in turn, caused reverse flow of the purge gas, activity could be lost from the trapping system. In fact, depressurization of the coolant system could also lead to loss of activity from the external traps. The temperature of the fuel material would not change appreciably during the accident conditions and, therefore, additional release of activity by diffusion would probably not be important. As in the case of the AGR or the EGCR, activity could be released from the fuel as a result of chemical reactions in the core. With the higher core temperatures in the Class C reactors, it is extremely important to design to prevent significant graphite oxidation, since such oxidation would not only lead to high temperatures in the core, but would also provide the mechanism for releasing activity from the fuel. The Peach Bottom reactor and the Dragon reactor designs include provisions to both prevent graphite oxidation and to control the release of activity that is postulated to occur at the time of a maximum credible accident.

The containment system for the Dragon reactor will be divided into two pressure vessels, designated the inner containment shell and the outer containment shell. The inner steel shell is designed to withstand a pressure of 10 psig and to have a leakage rate of about 0.1%/day. The outer containment shell will be built of concrete and is designed to withstand an internal pressure of 3 psig. The specified leakage rate for the outer containment shell is about 5%/day per psi of pressure differential. All those parts of the reactor that will contain large amounts of activity will be located within the inner containment shell. Ventilation of the inner and outer containment shells will be provided to remove heat and to control the location of fission products. A flow of air or nitrogen

will be supplied to the inner containment shell, and, if activity levels inside the inner containment shell become excessive, the inlet and outlet valves will close automatically and seal the vessel. Air flow will be supplied to the outer containment shell during normal operation, but at the time of an accident the normal ventilation will be stopped and the outer containment shell will be sealed. Recirculation units that include fission-product traps will draw air from the space between the inner and outer containment shells, filter the air, and then discharge it into the outer containment shell. Access to the inner containment shell will be possible only when the reactor is shut down, and even then personnel will be required to use protective clothing and breathing apparatus. There will be normal access to the outer containment shell, as required, although there will usually be no more than 10 persons in the area.

There is one outer containment building for the HTGR, which consists of a steel shell designed to withstand an internal pressure of 8 psig at a temperature of 150°F. The leakage rate is to be 0.2%/day at 8 psig. This containment building is separated into three general areas: the main containment room, the air room, and the equipment cavities. The main containment room will be depleted of oxygen by burning a hydrocarbon fuel in air to form CO_2 and water. The water will be condensed, and the CO_2 fed into the main containment room. Certain equipment that may require frequent maintenance will be located in the air room. Fresh air will be supplied to the room and discharged to the stack. The air room will be maintained at about atmospheric pressure, whereas the main containment room will have a slightly negative pressure during normal

operation. The equipment cavities that contain the primary system and the helium purification system will be provided with independent recirculation units for heat removal. In the event of an emergency, all discharge of ventilation air to the stack will be halted as valves close to seal the containment rooms. Following isolation of the containment rooms, the contents of the rooms may be recirculated through an emergency filter system to reduce the concentrations of activity inside the building.

Class D: Ceramic Fuel with Ceramic Cladding. The Julich reactor utilizes ceramic fuel and ceramic cladding but is basically different from the Dragon and the Peach Bottom reactors. The Julich reactor is a pebble-bed reactor and, therefore, a sweep gas cannot be utilized as in the Class C reactors to control the location of the radioactivity. The fuel material, moderator, coolant, and temperatures in the Julich reactor are not greatly different from those in the Class C reactors, except that the Julich reactor has a higher gas outlet temperature. The fuel elements and moderator are combined in spheres. Any activity released from the fuel will either circulate in the coolant or deposit on surfaces in the coolant system. Several barriers will be utilized to contain or to confine the release of activity.

The fuel will consist of graphite spheres that contain a fuel charge in the form of uranium carbide particles in a graphite matrix. The fuel material will retain most of the fission-product activity, but some of the noble gases and volatile fission products will diffuse from the fuel material and escape into the gas stream. The amount of activity that will be released is estimated to be about 6×10^5 curies. Of this amount about 25% will deposit on surfaces. The remainder will be mostly noble-gas activity.

The entire primary coolant system will be contained in a shell designed for an internal pressure of 20 atm that will normally operate at 10.25 atm. Immediately surrounding the pressure shell there will be a sealing gas contained in an outer shell. The pressure of the sealing gas will be 10.45 atm, and therefore any leakage will be into the primary system. The outer shell is designed for an internal pressure of 11 atm. Relief valves will protect the shells from overpressurization.

A shielded container will enclose all contaminated circuits of the plant. This container is designed for a maximum internal pressure of 2 atm, and the allowable leakage rate is 2.5 vol %/day at 2 atm. Access into the shielded container will be through one of three personnel air locks or through a materials gate. A sprinkler system provided to limit the pressure rise in the shielded container would be put in operation automatically if the pressure rose to 1 atm. The maximum pressure for the worst condition is estimated to be about 1.6 atm.

A conventional building will enclose the shielded container. Therefore, any release of activity from the shielded container will be confined in the building until released from exhaust domes located at the top of the building (46 m). Estimates of radiological hazards were based on release at 15 m, however, because of the influence of the local forest.

Class E: Ceramic Fuel with no Cladding. The Ultra High Temperature Reactor Experiment (UHTREX) will consist of a high-temperature, helium-cooled, graphite-moderated reactor with unclad fuel elements. The purpose of the experiment will be to evaluate the advantages of a simple fuel relative to the disadvantages that may result from high contamination in the coolant loop. Therefore the design is not predicated on the

basis of retaining or minimizing the release of activity from the fuel material. Fuel will be introduced into the graphite matrix by impegnation, and there will be no cladding on the fuel elements.

The containment barriers for the UHTREX will be the primary system and the containment building. The containment building will be a concrete structure with a steel liner designed for an internal pressure of 7.5 psig and a leakage rate of 1.0%/day. No special provision will be made to prevent chemical reaction between the graphite and the air inside the containment system following a depressurization accident. The containment system is designed for the pressure resulting from combustion in the core. It is assumed that in the event of the maximum credible accident, 100% of the noble gases, 10% of the iodine, and 1% of the other fission products will be released to the containment building. This would be a release of about 8×10^6 curies of activity.

Conclusions

The containment systems for gas-cooled reactors vary in concept from the standard pressure containment vessels presently utilized for water-cooled reactors because of the basic difference in the integrity of the fuel material and cladding during accident conditions. The low-power densities and high-heat capacities of gas-cooled reactors allow sufficient time for backup systems to become operative before significant fuel failures occur. The reactors with metal-clad metallic fuel, as well as those with metal-clad ceramic fuel, can be designed and operated so that only a small fraction of the activity in the fuel would be released in the event of an accident. Gas-cooled reactors of advanced types (Classes

C and D) have all the inherently safe characteristics of Class A and Class B reactors and are potentially capable either of retaining large fractions of the fission products or of controlling the location of the fission products. The containment requirements for these reactors appear to be more closely associated with those for Class A and Class B reactors than with those for water-cooled reactors. Although the more advanced gas-cooled reactors are designed to utilize containment structures that are capable of withstanding a pressure release, as well as the activity release, it is anticipated that operating experience with these reactors may ultimately lead to the conclusion that the pressure type of containment system is not required.

The gas-cooled reactor containment system that is based on retention of activity in the core relies heavily on the operation of equipment for long periods of time after an accident. The concept of dependence on operator action and the functioning of equipment varies in degree from that presently utilized for water-cooled reactors. The time available for taking action is sufficiently great, however, to afford reasonable assurance that in a carefully designed reactor, safety can be met by operator action. Furthermore the separation in time of the pressure release and any large activity release is such that, instead of the conventional pressure-containment vessel, additional containment can be provided in the form of a conventional structure that confines activity after the initial pressure release is relieved. Although insufficient information is available to justify the conclusion that future gas-cooled reactors will not need pressure containment, the basic mechanisms for fuel failure and the inherent features of gas-cooled reactors suggest that adequate

containment systems for gas-cooled reactors will rely primarily on the fuel material and to a lesser degree on some basic variation of either a pressure-relief or pressure-suppression containment or confinement system.

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V. CONSEQUENCES OF ACTIVITY RELEASE

NUCLEAR SAFETY IN FALLOUT SITUATIONS

Gordon M. Dunning*

The subject of nuclear safety in fallout situations might be covered in many ways, including a routine recounting of facts and figures. This article, however, describes the sequence of events and the reactions of the people involved in three incidents in which countermeasures were used in relation to fallout. One of these incidents required evacuation of 236 persons for periods of up to more than three years, in another some 4500 persons were asked to remain indoors for about 2 hr, and the third involved actions taken to reduce the intake of radioiodine by 200,000 persons. None of these stories or the relevant data are new, but they bear retelling to illustrate the types of situations that can develop, to record the decisions that were reached, and to outline the bases for these decisions. Emergency situations can become quite personal, and it will be evident in the following that the decisions were based not only on technical data but also on human judgment.

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The Pacific Incident

A kiloton shot, designated Bravo, was detonated on March 1, 1954 on a reef extending from the island of Namu on the northwestern part of Bikini Atoll. A map of the Marshall Islands showing the important atolls is presented in Fig. V-1. The night before the shot, a portion of the test team sailed from the test site on the USMS Ainsworth and, at shot time in the morning, stood off some 35 miles from ground zero. There came the usual countdown and then the intense flare of light. As the light dimmed sufficiently for dark glasses to be removed, there was the spectacle of the cloud as it rose to the tropopause, at about 55,000 ft and to its ultimate height of 114,000 ft.

It had been planned for the USMS Ainsworth to return the test team to the advance camp near the test site, but it was not long before the radiation monitors caused a change in the plans. The personnel went below deck for several hours to reduce the radiation exposures to minimal amounts (e.g., 150 mr), while the ship was returning to the base on Parry Island, Eniwetok. The shot had turned out to be of greater force and to have more fallout than expected.

Evacuation from Rongelap and Sifo Islands. At H (the hour of detonation) plus 31 hr, an aerial survey plane flew, as planned, over Rongelap Island, 115 statute miles to the east of ground zero, and reported activity of about 4 r/hr. The aerial reading subsequently was shown to be somewhat high (the radiation level 3 ft above the ground at the time of evacuation was about 1.5 m/hr at H plus 51 hr), but it triggered a chain of actions that was desirable. It was learned later that shifting winds had caused the fallout pattern to move southward over Rongelap

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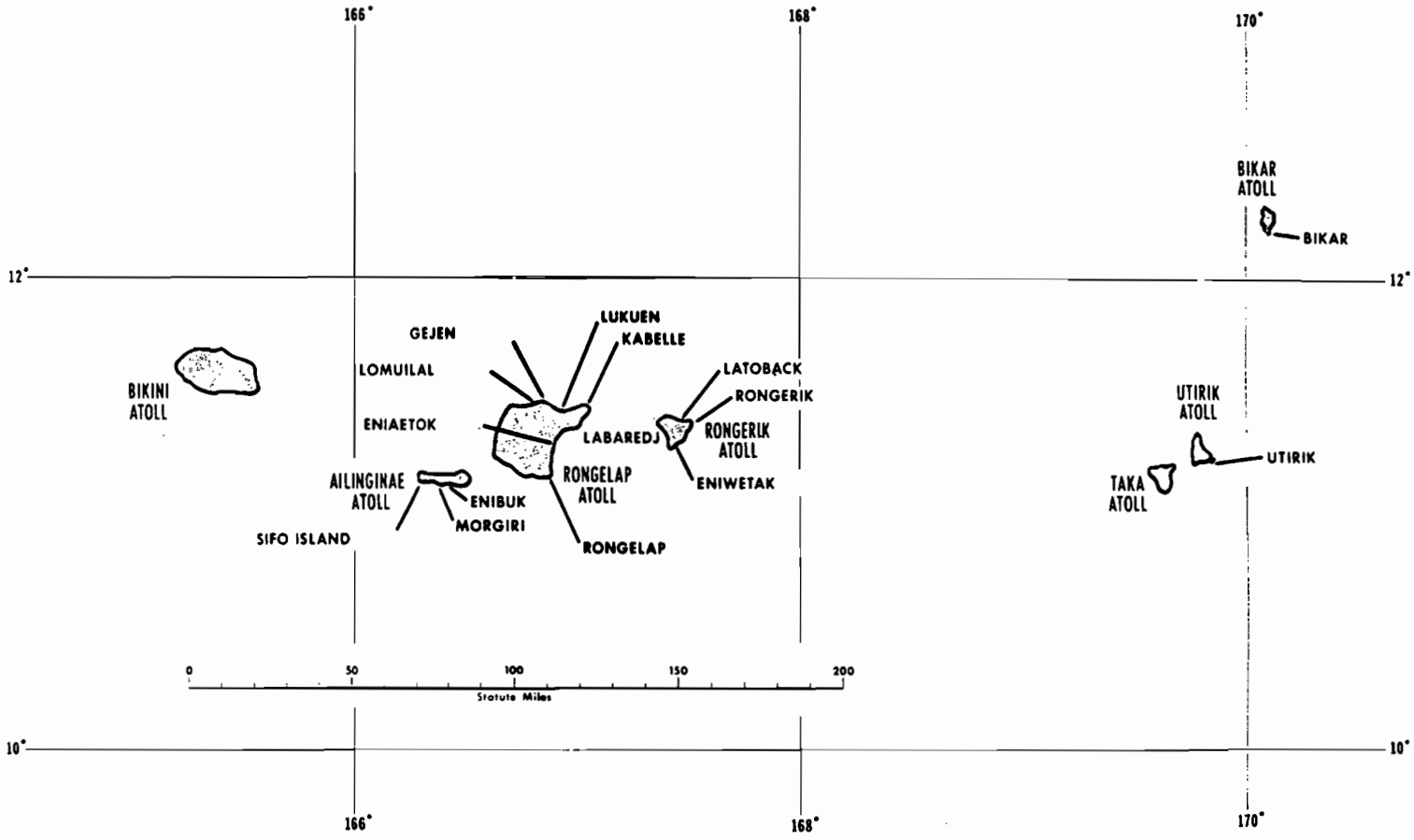

Fig. V-1. Map of Marshall Islands.

Island, Ailinginae Atoll, Rongerik Atoll and Utirik Atoll, rather than to the north, as predicted.

Although it had not been expected that evacuation would be necessary, plans for such an eventuality had been made, as they should be in a good safety plan. Both aircraft and surface ships were dispatched to Rongelap, and at about H plus 51 hr, 16 Rongelap Islanders were evacuated by air and 48 by surface ship. Although the radiation levels on the Island of Sifo on Ailinginae Atoll were less than one-half those on Rongelap Island, 18 inhabitants of this island were also evacuated by ship at about H plus 54 hr. They were all taken to Kwajalein and given the best possible medical care,[1] and their needs were amply supplied. They were moved to the Island of Ejit on Majuro Atoll in June 1954 and returned to their home islands on June 29, 1957.

Once the levels of radiation had been determined, the capabilities for evacuation had been verified, and the feasibility of evacuation had been established in terms of minimum hardship to the evacuees, it was not too difficult a decision to make to evacuate Rongelap and Sifo islands. The next decision was, however, more difficult to make.

Evacuation of Utirik. By late in the evening of the second day, radiation reports had been received from the island of Utirik, about 315 statute miles to the east. In discussions leading to the evacuation of the people of Utirik, the arguments against evacuation were that their lifetime exposure was estimated to be probably not more than 60 r if they remained on the island indefinitely, that evacuation would involve some 150 inhabitants, and that the evacuation would be a hardship to them. Arguments for evacuation were that there were ships capable of removing

the Utirik inhabitants by the third day after shot day, removal would save them possibly 45 r of exposure, and the major decision had already been made when the first Rongelap Islanders were evacuated. After weighing these factors, it was decided that the Utirik inhabitants should be removed.

The 154 Utirik people were evacuated at about H plus 78 hr to Kwajalein and given the same care as those from the Rongelap and Sifo Islands. They were returned to their home island of Utirik on June 5, 1954. When these matters were discussed later in the Trusteeship Council of the United Nations, it was possible to show that every possible action had been taken to protect the inhabitants of Utirik, even though there might have been some question of necessity for doing so.

Return of the Rongelap Islanders. The people from Rongelap and Sifo Islands were returned to their home island on June 29, 1957, over three years after their evacuation. The data and their interpretation that led to the decision for the return of these people to their homes are summarized below. The external gamma dose rates on Rongelap Island at the time of their return were:[2]

	Dose Rate (mr/hr)
Highest	0.13
Lowest	0.01
Average	0.03

One of the principal sets of criteria for whole-body radiation of a general population was stated in 1957 by the National Committee on Radiation Protection:[3]

> "The maximum permissible dose to the gonades for the population of the United States as a whole from all sources of background, shall not exceed 14 million rems per million of population over the period from conception up to age 30, and one-third that amount in each decade thereafter. Averaging should be done for the population group in which cross-breeding may be expected."

The radiation levels on Rongelap Island met these criteria.

The other principal factor was Sr^{90} in the bones. The most definitive statement on record in 1957 concerning strontium levels was that of the National Academy of Sciences:[4] "There seems no reason to hesitate to allow a universal human strontium-burden of 1/10 of the permissible...," that is, 100 strontium units.

Based on analyses of considerable data on environmental contamination on Rongelap Island and of the proposed countermeasures (such as supplementing the diet with imported foods), the author predicted in 1957 that "if land crabs are eliminated from their diet, the estimated future body burden of the Rongelapese would be substantially less than 100 micro-microcuries of Sr^{90} per gram of calcium," and as a double safety check "By means of the continuing medical examinations...it would be possible to note any tendency of untoward accumulation of strontium-90 with time, and appropriate action could be taken before excessive levels were reached."[2]

Conard of the Brookhaven National Laboratory has been following the medical history of the Rongelap Islanders in annual trips to the island.[5] The estimated body burdens through 1961 are summarized in Fig. V-2. The extreme difficulty of obtaining appropriate samples from small children almost precludes any direct measurements for the age group of most interest. The mean body burden for all samples (estimated from urinalysis) was

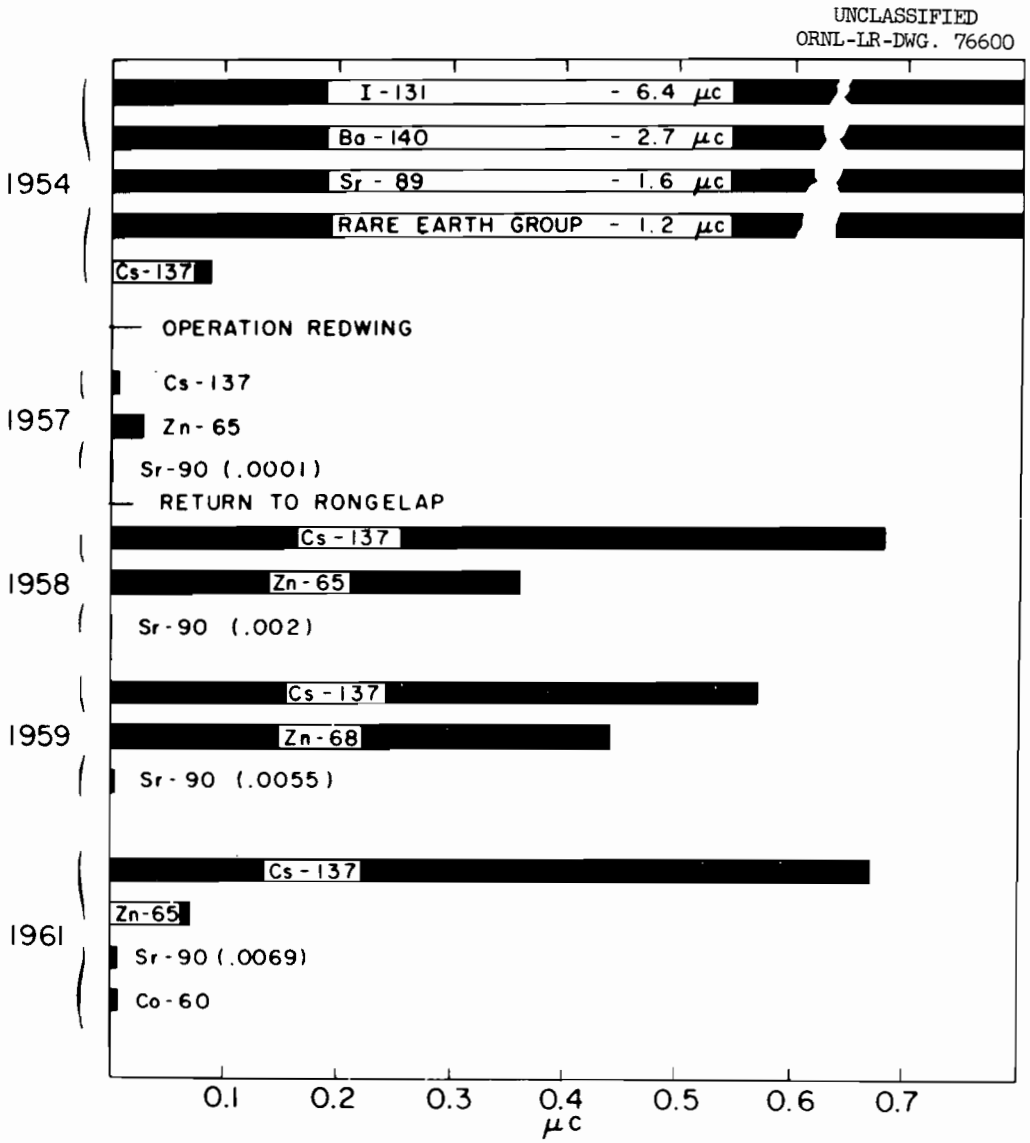

Fig. V-2. Estimated Body Burden of Radioisotopes in Rongelap People for the Period 1954 Through 1961.

6900 μμc. If this were an adult value, it would represent 6.9 strontium units. Other data indicate that the body burdens (in strontium units) of young children are higher than those of adults by a factor of 5 or more. The average Sr^{90} excretion by the Rongelap children was 159 μμc per gram of calcium, as contrasted with 30.9 for the mean of the adult population. None of the available data permit a precise determination of the Sr^{90} burden of the children (or of the number of land crabs eaten per day), but it would appear to be less than 100 strontium units. Of course, as the children grow older their body burdens (in strontium units) should decrease.

The St. George Incident

A 32 kiloton shot, designated Harry, was detonated from a 300-ft tower at the Nevada Test Site on May 19, 1953. There was more than the usual concern regarding the fallout pattern because an earlier (April 25, 1953) shot in the same upshot-knothole series, a 43 kiloton shot from a 300-ft tower, had placed the heaviest fallout pattern yet experienced from a nuclear detonation at the Nevada Test Site to the east-southeast across the southernmost part of Utah, and it was advisable not to have fallout from Harry occur in the same area. Even so, the predicted 0.5-r line extended out 250 miles, the distance at which the estimated whole-body external dose would be 0.5 r if persons were to live in the area indefinately.

Preparation for the Shot. Mobile monitoring teams were dispatched during the day preceding the shot. As was customary the Liaison Officer of the Federal Aviation Agency attached to the Test Organization at the

Nevada Test Site directed the closure of air spaces for commercial aircraft from the site out to specified distances, roughly 50 to 70 miles (principally to avoid the possibility of the flash of the detonation temporarily blinding pilots), and certain air lanes were closed for specified distances, altitudes, and times (depending on the size and placement of the shot and the predicted path of the fallout). Cloud tracker aircraft of the Test Organization were in position, and helicopter crews were available for close-in terrain surveys and C-20 and C-47 crews were ready for more distant terrain surveys. The usual ground and aerial sweeps were made in the afternoon to assure that there were no unauthorized persons in the close-in areas in the direction of the fallout. The technical crews reported their readiness for all experimental work on-site and off-site, and 0505 Pacific Daylight time, on the morning of May 19, 1953, the shot was made. Within a short time the technical success of the shot was determined, and the technical crews went back to camp.

Roadblocks. The first on-site and off-site reports were encouraging. The fallout was progressing to the east-northeast, and it crossed U. S. Highway 93 south of Alamo and north of Glendale Junction, Nevada, as predicted. In anticipation of this event, roadblocks were established on Highway 93 at Alamo at 0715 and at Glendale Junction at 0725 (see Fig. V-3). This prevented persons from being directly in the fallout as it occurred and reduced the whole-body exposures and the direct contamination of personnel and equipment. The roadblocks were removed at 0851, and cars were monitored after they had traveled through the area. Precautionary closing of Highway 91–93 between Las Vegas and Glendale Junction

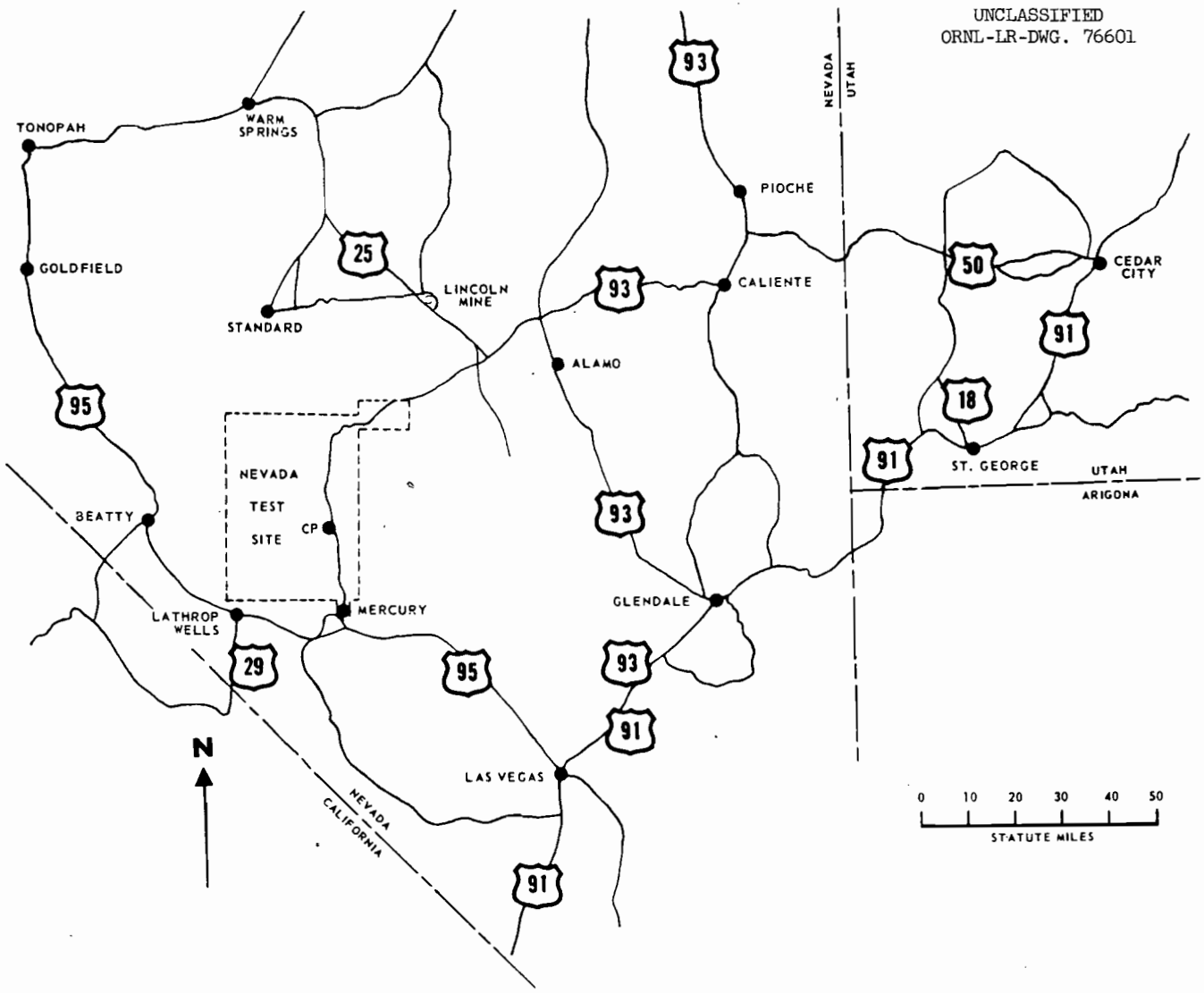

Fig. V-3. Nevada Test Site and Environs.

was ordered at 0735 and lifted at 0805. A precautionary roadblock was established at St. George at 0745, but it was not until 1130 that this roadblock was lifted. All in all, hundreds of cars were monitored and about 40 to 50 vehicles were washed (at Government expense) according to the established criteria.[6]

Restrictions at Groom Mine. Groom Mine was not directly in the path of the predicted fallout, but since it was the nearest inhabited place, about 30 miles from ground zero toward Alamo, monitors were stationed there. At 0622 the radiation level rose rapidly to 140 mr/hr, and the few inhabitants were asked to remain indoors. They were released at 0748 after the cloud had passed and the radiation levels had subsided. At 0920 the radiation levels outside were 11 mr/hr and dropping.

The trajectory of the air mass containing the radioactive debris passed south of Groom Mine, moved in an east-northeasterly direction, and crossed Highway 93, south of Alamo, all about as predicted. The monitoring data suggested that the pattern was somewhat farther south than predicted, but not disturbingly so, since beyond Highway 93 and in the line of the trajectory there was uninhabited country for many miles.

Curfew at St. George. The monitors at the St. George roadblock (actually at the junction of Highways 91 and 18 to the west of St. George) noted that at 0845 the radiation levels on the cars being monitored were increasing, as were the levels in the environment. By 0910 the levels had risen to 320 mr/hr and a quick check of an automatic background recorder at nearby Dixie College verified that this was not failure of the field survey instruments. The head of the monitoring team then called the Control Point for instructions. This was the first indication at

the Control Point that the winds had carried the fallout far enough southward to create a problem.

In view of the fallout pattern, a decision was made to ask the residents of St. George to stay indoors from about 0930 to 1130. There were about 4500 persons involved, who were spread throughout the city. In making the decision, consideration was given to whether a harmful panic might occur if an emergency order was issued. The principal question was, however, whether the radiation level was high enough to warrant an emergency directive. It could be argued that the potential radiation dose, which actually turned out to be about 2 r, was not sufficient to take the risk of the possibily greater hazard of a panic. There had been an unexpected shift in the winds, however, and it was not known at that time how much higher the radiation levels might go. The decision was therefore made to ask the people of St. George to go indoors.

As reported later, "At 0925 instructions were received to have the people in St. George take cover. The sheriff was notified and he in turn contacted the radio station in Cedar City to get the announcement over the air. In addition, the school principals were notified of the situation so that the children would not be sent out into the open during recess periods. At 0940 the bulk of the population in the city of St. George was under cover. The effectiveness of the operation was amazing."

A key statement here is that "The Sheriff was notified... ." It was not the AEC that directed the people of St. George to stay indoors. It was a recognized officer of the law and a local man whom everyone knew. But even this might not have been enough, if it had not been for the careful planning and the orientation of the Sheriff and his deputies beforehand.

They knew what to do and why before the emergency arose. The incident also showed the value of a well-informed public. Programs of public information and education were re-emphasized after this event.

The Salt Lake City Incident

Four low-yield nuclear devices were detonated on or near the surface at the Nevada Test Site on July 7, 11, 14, and 17, 1962. One deep underground shot (Sedan, a part of the Plowshare Program) was made on July 6, with a total yield of about 100 kilotons, a fission yield of less than 30%, and about 95% of the radioactivity trapped in the ground.[7]

Each of these detonations had been given close scrutiny, and the controlling radiological safety factor was the potential external whole-body exposures to nearby off-site populations; that is, this requirement had to be satisfactorily met or the shot could not be made. The great care exercised in the conduct of the activities at the Nevada Test Site is reflected in the record of exposure to off-site personnel. From the start of the current nuclear tests in September 1961 to one year later, the highest individual whole-body exposure off-site was about 0.5 r.

Iodine-131 in Milk. With increased alertness to possible concentrations of I^{131} in milk and the increased ease and swiftness of monitoring by means of gamma-ray spectrometry, the rise of I^{131} levels in milk in the Salt Lake City area was noted following the Sedan shot, as shown in Table V-1. As had been noted on other occasions, a few days passed between the time of deposition and the peak value of the iodine activity in the milk. The initial rise in the activity in July at Salt Lake City probably was due to fallout from the Sedan shot, and the principal

Table V-1. IODINE ACTIVITY IN COMPOSITE SAMPLES OF SALT LAKE CITY MILK

Date Sample Was Collected	I^{131} Content[a] (μμc/liter)
June 20, 1962	Not detectable
July 3	Not detectable
6	Not detectable
10	20
11	60
12	30
13	300
14	250
15	390
16	410
17	160
18	450
20	1660
21	1190
22	450 } from different sampling points
22	1390 } from different sampling points
25	2050
26	1960
27	1290
28	960
29	730
31	570
August 2	640
3	590
4	370
5	410
6	520
7	320
8	470
10	210
11	250
12	250
13	550
14	200
15	90
16	50
17	170
20	110
21	110
22	100

[a]Precise relationships of activity versus time could not be established because some cows were placed on dry feed and then returned to pasture at different times.

contamination that followed later probably resulted from subsequent shots on July 11 and 14.

There was considerable public reaction in the Salt Lake City area, because the newspaper and TV stories carrying the values of concentrations of I^{131} in the milk, which were essentially meaningless to the layman, also carried statements relating them to the Federal Radiation Council Guides. The 2050 μμc of I^{131} per liter of milk did seem alarming when compared with the permissible 100 of the Federal Radiation Council Guide.[8,9] Strong pressure built up for action on the part of public officials. The State and City health authorities met with representatives of the milk industry, and as a consequence several actions were taken by the milk industry, as announced on August 8, 1962. Of the 759 milk producers in the Salt Lake City area, 285 put their cows on dry feed, 211 others diverted their milk into manufactured milk. This represented 53,188 gal of the 77,277 gallon total daily milk production.

Obviously these were not minimal actions, and therefore it is important to consider whether these countermeasures were necessary. It can be argued that, since all radiation is bad, anything and everything that can be done to reduce it is good. On the other hand, removing the milk supply of children may be worse than accepting radiation exposures in the ranges experienced from fallout. As the Federal Radiation Council stated in its letter to the Joint Committee on Atomic Energy on August 17, 1962, "...radiation exposures anywhere near the Guides involve risks so slight that countermeasures which themselves involve any slight hazard may have a net adverse rather than [a] favorable effect."

In relation to the countermeasures taken in Utah and similar actions in Minnesota, the Federal Radiation Council sent a second letter to the Joint Committee on Atomic Energy, that stated the case even more specifically:

> "The Council recognizes that premature action has been taken in some areas to reduce the intake of iodine-131 which action the Council would not have recommended under its interpretation of the Guides. The Council would like to emphasize this point by reiteration of its position outlined in its letter of August 17 to you. There, we stated that '...there has been widespread misunderstanding concerning these Guides... .' We want to point out that they '...are not intended to set a line at which protective action should be taken or to indicate what kind of action should be taken.'"[10]

Interpretation of Federal Radiation Council Guides. The real problem at Salt Lake City was that the countermeasures were based on a misunderstanding of the meaning and intended uses of the Federal Radiation Council Guides. This emphasizes the responsibilities of scientists not only to establish radiation protection criteria but also to ensure their proper interpretation. One of the principal points of misunderstanding in Salt Lake City was the meaning of the 36,500 μμc of I^{133} given in the guide. The guide establishes a permissible dose of 1500 mrem per year to the thyroid, if the measurements are made on individuals, and 500 mrem per year, if "suitable sample" of the population is measured.[8] According to a series of assumptions, including the assumption that a child drinks a liter of milk a day, about 100 μμc of I^{131} per liter would give a 500 mrem dose to the thyroid in a year and 100 μμc per day would add up to 36,500 μμc in a year. There are several things that can be said about these numbers.

1. The Federal Radiation Council Guides are based on a lifetime of exposure and it is only for administrative and regulatory purposes that

it was decided to average the intake over one year's time. Even this means that the Guides permit 36,500 μμc intake per year for each and every year for a lifetime. Therefore there should not be an undue concern if the guide values are exceeded in some years, especially for I^{131}, which has a half-life of only 8 days.

2. The iodine intake limit established by the Guides is actually the intake per year for individuals in the population. Recognizing that the intake of all individuals cannot be properly monitored, the Guides state that if only a suitable sample of a population is monitored, a value of one-third the individual intake criterion should be used. The criterion for individuals of 109,500 μμc has relevance only when the individual farms that show the highest I^{131} content in the milk are considered.

3. The I^{131} criteria were established to limit the radiation dose to the most sensitive segment of the population, that is, young children. Limiting the exposure to 500 mrem per year to young children means that adults would receive only one-tenth of this dose, even if they ingested the same amount of I^{131}, since adults have larger thyroids.

Another area that needs a fuller explanation is the statement that the Federal Radiation Council numerical guides apply only to "normal peacetime operations." The necessity of balancing benefits against risks in setting radiation protection guides is automatically imposed when the assumption is made that there is no threshold of effects, that is, that each increment of dose above zero has its commensurate biological effects. This balancing was performed by the Federal Radiation Council for the anticipated benefits of "normal peacetime operations" compared with the

possible biological effects from the radiation doses. If a different atomic energy activity were undertaken, such as nuclear weapons testing, there would have to be an independent balancing of the benefits against its risks.

These and other points were brought on by the author at the Congressional hearings on radiation standards, including fallout, on June 7, 1962:[11,12]

> "In summary — the FRC guides, even in the proper context of normal peacetime operations only, do not, nor were they intended to, constitute precise health standards. They should not be likened to a precipice such as is implied when we speak of environmental levels approaching the top of Range II as though this should call for drastic countermeasures such as taking milk away from babies or disrupting normal dietary habits. The health hazards from such actions can outweigh the radiation exposures.
>
> "Finally, and most importantly, the FRC Guides were developed for, and should be applied only to, normal peacetime operations. There must be an independent balancing of the need (i.e., benefits) for nuclear weapons testing versus potential health risks from fallout. This should be clarified at once, before there is further confusion and before there may be an ill-advised action taken by some regulatory body."

Two Federal Radiation Council letters[10,13] now clarify the issue. Furthermore, it is Council policy to establish the intake and exposure levels at "...the lowest possible level at which it... [is] believed that nuclear industrial technology... [can] be developed."[13] In substantiation of this policy, the Federal Radiation Council arbitrarily lowered the recommended permissible daily intake values for Sr^{90} and Sr^{89} because "... the Council's study indicated that there is currently no known operational requirement for an intake value as high as the one corresponding to the Radiation Protection Guide."[8]

In terms of this discussion, the most important thing to be said about the Federal Radiation Council Guides is that they never were intended to be standards representing serious health hazards. The Federal Radiation Council Guides were developed to control the release of radioactivity into the environment from "normal peacetime operations." For example, if an atomic energy installation released radioactivity at levels in excess of those recommended, there would be evidence that improvements in the operation were needed, such as better filters in the stacks, improved radiation control, etc. Obviously, guide values developed primarily for use by industry in restricting the release of radioactivity to the general environment outside controlled areas must be materially lower than those that might constitute a serious health hazard.

The Federal Radiation Council's first letter[13] clarifies this point:

> "In summary then, the Guides are not intended to be a dividing line between safety and danger. We have assumed that there is some slight risk to health from any level of radiation exposure, however low, even at or below the low levels set by the Guides. At the same time we do not believe there is any risk of a major health hazard until exposure levels are many times above the Guide levels. For example, this is borne out in relation to iodine-131 by the report to the Federal Radiation Council of the National Academy of Sciences [ref 14]... ."

A summarizing statement in the National Academy of Science report concerning I^{131} was that "... There is no evidence on hand, except for one doubtful case in a child, that any of the treatments of hyperthyroidism has produced a thyroid cancer, although doses have ranged from a few thousand rads upward."

A dose of "a few thousand rads upward" compares with the Federal Radiation Council Guides for I^{131} as follows: if a man were conceived,

born, lived a full span of life in an environment where the amount of I^{131} remained constant at a level as high as the upper limit of Range II of the Federal Radiation Council Guides (i.e., an uptake of 100 μμc per day), the total 70-year radiation dose to his thyroid would be only about 5 r, which for the radiation being considered is approximately equivalent to 5 rads. In fact, based on the Federal Radiation Council estimates[15] of the dose to thyroids of children during the past years when testing took place, the calculated 70-year dose would be only 1 to 2 r, even if a major series of tests were held each year.

TRITIUM HAZARDS IN HEAVY-WATER-MODERATED REACTORS

H. L. Butler*

Tritium is produced in heavy-water-moderated reactors as a result of neutron capture by deuterium.[16] Although tritium is a very weak beta emitter (0.018 Mev maximum energy), it may, in the oxide form, constitute a serious hazard. Unlike most radioactive materials, tritium is assimilated not only by respiratory processes but to an equal degree by absorption through the skin. No more than 0.1% of the tritium assimilated as T_2 is retained, but essentially all that assimilated as the oxide remains in the body.[17] Safety controls are conservatively based on the premise that tritium is always encountered in the more hazardous form. The maximum permissible body burden of 1 mc recommended[18] by the National Committee on Radiological Protection is used as the basic exposure value. The NCRP also gives 2×10^{-5} $\mu c/cm^3$ in air as the mpc for a 40-hr work week and 5×10^{-6} $\mu c/cm^3$ for continuous occupational exposure.

The concentration of tritium in heavy water used as moderating material is a function of reactor power level and the irradiation time. A primary consideration in minimizing tritium exposure is containment of the heavy water, and containment becomes increasingly necessary as

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the tritium concentration increases. In heavy-water-moderated reactors, the heavy water may serve not only as a moderator but also as a coolant. The resultant complex and extensive system of heat exchangers and lines provides opportunities for line breaks and occasional spills. Repair and replacement of pumps, piping, and heat exchangers therefore consistently require consideration of tritium hazards. Other potential sources of tritium exposure include removal of fuel assemblies and other components that have been in contact with the moderator in a reactor, reprocessing of degraded moderator material, regeneration of ion-exchange resins, and laboratory analysis of moderator material.

As with other radioisotopes, the effective control of tritium exposure is a function of facility design, effectiveness of training programs, use of protective equipment, and adherence to established procedural controls. In addition the ability to detect and measure the amount of tritium present is essential, but difficult, because of the low energy of the tritium beta activity. Adequate techniques have been developed, however, for measuring concentrations in air, for determining surface contamination, and for performing bioassays.

Exposure-Control Practices

Facility Design. Concern for tritium hazards must be reflected in the design of reactor facilities. Process areas should be kept at a negative pressure relative to offices and operating areas, and the exposure potential should be further minimized by limiting the number of access points and providing high-flow ventilation systems.

Training. While a few investigators had carried out experiments with both animals and humans using tritium, there was very little industrial experience with this toxic material when the first reactor went critical at the Savannah River Plant in 1953.[19-21] It was found that early training covering the physical and radiological properties of tritium, surface contamination control, and use of protective clothing and protective techniques was necessary. As personnel gained experience, more detailed information could be provided on such matters as control limits, maximum permissible body burdens, and monitoring instruments. During the initial months of reactor operation, the tritium concentration in the moderator was relatively low, and thus an opportunity was provided to increase the proficiency of maintenance and production crews in the repair of coolant and moderator systems and of laboratory personnel in the analysis of tritiated moderator material while the tritium hazard was relatively small.

Protective Equipment. Respiratory and other protective devices constitute another facet of tritium exposure control, since tritium is assimilated both by inhalation and by absorption through the skin. As a rule, where tritium concentrations in air are below the limit recommended by the NCRP (5×10^{-6} $\mu c/cm^3$),[18] no protection is required. Exposure at the mpc level for 8 hr, assuming a breathing rate[22] of 20 liters/min and considering skin absorption, would result in an assimilation of approximately 10% of the maximum permissible body burden. As the concentrations increase, however, the need for protection must be considered.

During the initial months of reactor operation at the Savannah River Plant, it was found that a nomograph could be used to determine permissible working times in low levels of tritium without respiratory protection. As concentrations increased, air-supplied face masks were required to permit additional working time. Because of equal absorption through the skin, however, the face masks provided only twice as much working time, and it therefore became necessary to provide air-supplied suits for complete enclosure of the body. The magnitude of this problem at the Savannah River Plant is reflected in the fact that during the past three years at least one of every four air samples obtained in reactor maintenance areas (or locations where moderator material is handled) and measured for tritium had a concentration of 10^{-5} $\mu c/cm^3$ or higher. In 1959, the number of air samples containing 10^{-5} $\mu c/cm^3$ or higher was slightly over 4000; by 1961, this number had more than doubled, to approximately 8300. Concentrations of tritium in breathing-zone air ranged as high as 10^{-2} $\mu c/cm^3$.

After evaluation of a number of plastic materials for permeability,[23] launderability, and general physical characteristics,[24] a 6-mil-thick two-piece suit of polyvinyl chloride was adopted. The wearer can work in a suit of this type in atmospheric tritium concentrations up to 10^{-2} $\mu c/cm^3$ for periods up to 6 hr without exceeding a 1-mc assimilation. Futhermore, no loss of protection has been found to result from laundering the suits for reuse. Data were initially obtained from tests using an air flow of 6 to 8 cfm through the suits. It was found, however, that an air supply of 10 to 15 cfm provided greater employee comfort. The plastic suit provides complete body enclosure and thus gives protection

not only from airborne tritium but also from overhead leakage, spills, and splashes.

Rubber gloves and overshoes are also important items of protective clothing. They, too, can be reused after laundering. On most jobs red rubber gloves are adequate when plastic suits are worn, but occasionally as many as three pairs of thin rubber gloves may be required. Experience has shown that where highly contaminated equipment is handled, the outer gloves should be removed and replaced frequently in order to minimize assimilation.

Work Authorization. The most effective means of personnel protection is the preplanning of work. Those who are to do the work and representatives of the health physics organization should participate in the preplanning sessions. Agreement must be reached on the degree of protection required based on the known hazard. Protective clothing, time limits, and monitoring requirements should be defined and agreed to.

Detection and Measurement

Airborne Tritium. Air monitoring, although essential in the control of tritium hazards, is difficult because of the low energy of the tritium beta activity (average energy of only 6 kev). Three instruments have been found that meet most requirements: a Kanne chamber, a portable tritium monitor ("sniffer"), and a small ionization chamber. The Kanne chamber[25,26] is used where continuous monitoring is required. Air is drawn through a large ionization chamber (51.5 liters) at the rate of 3 to 4 cfm using an air pump. Tritium concentrations in the range 10^{-5} to 5 $\mu c/cm^3$ can be measured. Current is measured by a micromicroammeter, and the amplifiers

are logarithmic. The chambers are located where low background radiation levels exist, since chamber response to radiation fields above 1 mr/hr is significant. Mobile Kanne units can be provided, in addition to stationary units. Contamination of the chambers is usually not burdensome; if it does occur, purging with heated tritium-free air effects decontamination. Interference from Ar^{41} or other gaseous radioactivity must be considered. Half-life determinations of air isolated in the chamber will show the presence of Ar^{41} or other radionuclides.

The portable tritium monitor[27] is based on the Los Alamos "sniffer"[28] but has a compensation chamber that allows tritium measurements in uniform gamma fields up to 100 mr/hr. Its range is 1 to 2500×10^{-5} $\mu c/cm^3$, and its weight is 16 lb.

In locations that cannot be sampled with either the Kanne chamber or the portable tritium monitor, smaller ionization chambers can be used for "grab" samples. Chambers of 1-liter capacity have been found to be satisfactory. The chamber is evacuated prior to its use in the location to be sampled and should be returned to a low-background area for measurement of ionization, using a vibrating-reed electrometer.

Controls for tritium in air are interrelated with bioassay and annual dose limits. A maximum permissible body burden of 1 mc has been found to be a workable standard at the Savannah River Plant. Based on a total-body-fluids volume of 43.4 liters,[17] this is equivalent to 23 μc of tritium per liter. For control purposes, a 20 μc/liter value has been used; personnel who assimilate tritium above this level are removed from all work where further uptakes of tritium may occur. When the assimilated tritium exceeds 40 μc/liter, the employee is removed from all work related to radiation.

In either event, the employee cannot return for further work in tritium atmospheres until the body buden is 10 μc/liter or below. Good control has been realized by requiring personnel working in tritiated atmospheres to wear plastic suits for maintenance work on line breaks or when the airborne tritium concentration exceeds 20×10^{-5} μc/cm^3.

Surface Contamination. The lack of portable instruments that can detect the very-low-energy tritium beta activity complicates survey procedures for surface contamination. The need for measurement is obviated somewhat by the fact that tritium contamination is usually accompanied by fission-product contamination. Detection in this case is accomplished with conventional beta-gamma detectors; and, since the limits for permissible fission-product contamination are low, these are the controlling factors. Normally, successful cleaning to remove the fission products removes the tritium as well. When no beta-gamma contamination exists, tritium must be measured using the conventional smear technique. Kraft paper disks 3 cm in diameter can be used, and an area of about 1 ft^2 should be smeared. These disks can be counted using proportional gas-flow counters in either a methane or a "Q" gas atmosphere ("Q" gas is a mixture of 98% helium and 2% isobutane). If the counter background is approximately 200 counts/min, the minimum detectable contamination is 100 counts/min. Surface contamination of 100,000 counts/min and higher may exist without need for respiratory protection if there is adequate ventilation.[29,30] In contrast, the enclosure of only a small piece of equipment, contaminated to a lesser extent, may present a serious breathing hazard when the packaged item is opened in a poorly ventilated area. In such cases, the tritium concentration in air may approximate 1 μc/cm^3

momentarily as a result of outgassing from the contaminated surfaces. Ventilation is therefore an important factor in surface-contamination considerations. Where feasible, work that may involve a large amount of contamination should be performed in a well-vented hood. The probability of tritium assimilation from surface contamination is small in well-ventilated areas. The values given below represent prudent and practical limits:

	Limits for Tritium Surface Contamination (counts/min per item or per ft^2)
Clean areas	200*
Regulated areas	500
Regulated-area tools and equipment	500
Radiation danger zone tools	200,000
Laundered rubber garments	1,000
Laundered plastic suits	
Outside	2,000
Inside	1,000
Sterilized face masks	
Outside	2,000
Inside	500

Bio-Assay. The concentration of tritium in air can be measured and permissible working times can be established, but it is only through bio-assay of urine that the effectiveness of these controls can be accurately determined. Within 2 hr after exposure, tritium reaches equilibrium with body fluids. Urine samples of workers whose work assignments have entailed potential exposure to tritium should therefore be analyzed daily. In this

*An area is considered to be clean if the contamination is 200 counts/min or one-fourth the counter background, whichever is less.

way, personnel with significant assimilations can be withheld from further exposure.

Liquid scintillation counters have been found most effective for these analyses.[31] Where assimilations exceed 20 μc/liter, the effective half life should be determined and used in the dose calculations. In calculations for uptakes below this level, the biological half life of 12 days can be used.

Tritium assimilations should be considered as whole-body irradiation, and the dose from tritium should be added to that received from gamma radiation. When bioassay data indicate an uptake of 5 μc/liter or greater at the Savannah River Plant, the dose from tritium is promptly added to the employee's radiation exposure record. (With an assimilation of 5 μc/liter and a 12-day half life, the dose is 45 mrem.[32]) At least twice a year, the dose from chronic uptake of less than 5 μc/liter is added to the record.

Environmental Monitoring. A continuous monitoring program is used to measure tritium, as well as the other radioisotopes, that might be released. Both air and rainwater samples are collected continuously at monitoring stations situated to detect significant releases regardless of wind direction. It is important to establish monitoring sites not only at locations near the potential source of release but also at locations which reflect the hazard to the population at large. Twenty sampling stations are provided over a radius of 25 miles for sampling tritium in air at the Savannah River Plant.[33]

The removal of spent fuel elements from a reactor results in the transfer of moderator material containing tritium. The fact that this

might be released to streams, and the ever-present possibility of leakage, make sampling of Plant effluents and streams an important phase of environmental control.[34]

ESTIMATION OF MPC VALUES FOR MIXTURES OF RADIONUCLIDES

(Held over from Vol. 4, No. 2)

SELECTION OF SAMPLES FOR AREA MONITORING

(Held over from Vol. 4, No. 2)

RUTHENIUM: A SPECIAL SAFETY PROBLEM

(Held over from Vol. 4, No. 2)

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VI. CURRENT EVENTS

HOMOGENEOUS REACTOR EXPERIMENT NO. 2
OPERATING EXPERIENCE

The Homogeneous Reactor Experiment No. 2 was terminated in May 1961. The HRE-2 was an aqueous homogeneous reactor in which the UO_2SO_4-$CuSO_4$-H_2SO_4 fuel solution was circulated through an external heat exchanger to produce steam.[1,2] It was built by the Oak Ridge National Laboratory to advance the technology of the two-region breeder concept, and it had a 32-in.-diam core separated from a blanket region by a thin Zircaloy-2 core tank. The core and blanket circulating loops operated at high pressure (1000 to 1800 psig), and the solution temperatures were normally between 250 and 280°C. The design thermal power was 5 Mw.

There were some unusual nuclear safety aspects of HRE-2 operation because of the fluid state of the fuel and the absence of the usual nuclear control elements. Instead of shim rods, the fuel concentration was used to control the average temperature. The most rapid shutdown or "scram" possible consisted of draining the fuel solution into dump tanks, a relatively slow process that required between 5 and 10 min. Reliance for power regulation was placed solely on the strong negative temperature coefficient of reactivity. Because the fuel was fluid and at high pressure, there were also unusual problems in containment of the radioactivity. One of the major objectives of the experiment was the demonstration that containment was feasible. Although the HRE-2 containment system was designed before present-day concepts were developed, stringent requirements were applied that compare favorably with current practices in reactor technology. As originally designed, at least two barriers prevented release of radioactivity to the atmosphere; namely, the primary

system and the equipment cell. Although the design proved successful in preventing all but very minor releases, additional protection was provided by changes made during the years of operation.

Nuclear operation of the HRE-2 began in December 1957, with D_2O in the blanket region. After April 1958, when a hole developed in the tank separating the core and blanket, the reactor was operated with fuel solution in both regions. In January 1960, after 105 days of continuous operation, a second hole was melted in the core tank.[3] In the following nine-month period, the holes were patched, diffuser screens in the core were removed, and the direction of flow through the core was reversed to go from top to bottom. The latter two changes were for the purpose of eliminating hot, stagnant regions and preventing heavy-phase accumulation in the core. These undesirable conditions were related to the separation of uranium from the fuel solution that had been observed under some conditions when the reactor was operated at high power. The separation of the uranium from the fuel solution had caused some reactivity disturbances and the core-tank damage. The experimental program was devoted largely to the investigation of fuel-solution stability under various conditions, concurrent with the demonstration of system and component reliability, containment, feasibility of maintenance, and continuous removal of fission and corrosion products.[4]

In the spring of 1961, a decision was reached to discontinue large-scale work on aqueous homogeneous reactors at ORNL, and, on May 1, operation of HRE-2 was terminated. During the 40 months following initial criticality, the reactor had been critical for 8840 hr and had produced heat equivalent to 135 days at full power.

Protection Against Damaging Reactivity Excursions

The feasibility of building a reactor without fast-acting nuclear control elements depends upon the nuclear stability of the reactor; that is, it must not be possible for a small distrubance to lead to divergent power oscillations and consequent damage. The characteristics of the HRE-2 were such that it was quite stable, even when subjected to many types of disturbances. This had been predicted on the basis of kinetics calculations and HRE-1 experience.[5] The operation of the HRE-2 was reviewed from the standpoint of nuclear safety a number of times, and modifications were made in the safety circuits to accommodate changes in the conditions of the reactor and the manner of operation.

Compensation for reactivity additions in the HRE-2 was through the temperature coefficient of reactivity, which was negative at all operating temperatues and ranged from -0.9×10^{-3} $\Delta k/°C$ at 20°C to -3.3×10^{-3} $\Delta k/°C$ at 280°C. The large effect of temperature on reactivity was principally the result of expansion of the fuel solution in the core and blanket. Such expansion, if it occurred in a short time, ejected fluid from the active regions of the core and blanket into the closely coupled, steam-filled pressurizers. (Expansion over a longer period of time was accommodated by automatic letdown of solution into the dump tanks.) Rupture of the core tank or reactor pressure vessel by a pressure surge that caused the rapid ejection of solution, while improbable, would have been the most likely form of damage from a reactivity excursion.

In a reactor, such as the HRE-2, with a high operating temperature, circulating fuel and moderator, and a strong temperature coefficient of reactivity, the potential reactivity additions are quite large. Operation

that would result in excessive reactivity additions was prevented in the HRE-2 by interlocks or, where possible, by restrictions on the design of the physical system.[6] Restrictions were determined from the maximum permissible pressure rise and the relationships between the design parameters, reactivity additions, and pressure surges. An illustration was the limitation of the fuel circulation rate when the fuel-circulating pump was started. With the pump off, the solution in the heat exchanger tubes might, as a result of abnormal operation, have become cooled to well below the core temperature, and a sudden startup could have injected cool fluid and increased reactivity at rates up to 1.7% $(\Delta k/k)$/sec. Because the resulting pressure surge would have caused undesirably high stresses in the core tank, the rate of possible reactivity addition was limited by automatic rotation of the centrifugal pump in the reverse direction (resulting in flow only one-third of normal) during the first 45 sec of circulation.

In addition to the example given above, there were a number of other interlocks that guarded against reactivity additions by injection of unusually concentrated or cool fluid. There were no interlocks actuated solely by reactor neutron level or period, because there was no external control that could have acted to stop a rapid nuclear excursion once it had been initiated. (There was an interlock that would close the steam withdrawal valves in case both the neutron level and the heat exchanger temperatures indicated that the core power was 40% above normal.) All control circuits were displayed schematically on a board in the main reactor control room, and interlocks were never jumpered, except by the use of phone jacks inserted in this board, a policy which made very

obvious any temporary changes in the circuits. (The use of jumpers had to be approved by the Operations Chief, and normally the board was clear.)

After the first hole developed in the core tank it became necessary to operate with fuel solution in both the core and the blanket. Further, one of the pressurizers had to be filled with liquid because the two pressurizer levels could not be controlled separately. Calculations showed that the core pressurizer alone would provide adequate surge space to prevent overpressurization of the pressure vessel if the blanket concentration were kept below 60% of that in the core. Operation was therefore resumed with only the one pressurizer, but because of the importance of ensuring against a pressurizer-level-detector failure, the installed float-type level element was supplemented by an independent level detector. This consisted simply of an alarm actuated by a thermocouple that would sense the rise of liquid into the pressurizer. In addition, a routine procedure was instituted for daily checking the response of both pressurizer-level-detecting devices.

Other changes made for operation with a single pressurizer included daily checks of the pressure-instrument response and a reduced loading pressure on the pneumatic operators of the dump valves. Part of the safety circuits, from the start, was an interlock that opened the dump valves automatically if the indicated reactor pressure rose 800 psi above normal. (The pressurizers were vented automatically at a pressure of 500 psi above normal.) The daily check was to show that the signal to this interlock was "alive." The dump-valve-operator loading pressure was reduced to permit the dump valves to act as pressure-relief valves when the fuel pressure under the seat of the valves rose to about 2500 psig.

After the core-tank holes were patched, operation with both pressurizers was resumed. Inasmuch as the failure of only one pressurizer would not endanger the pressure vessel and because of increasing confidence in the pressurizer instrumentation, the daily checks of level- and pressure-instrument response were discontinued, and the loading air on the dump valves was restored to the original pressure.

A problem of nuclear safety arose during reactor operation that was not susceptible to control by interlocks. Solution inventories and reactivity changes showed that, under some conditions, uranium separated from the solution in a reversible, power-dependent manner. Small amounts of this uranium, which apparently deposited in the lower part of the core, occasionally shifted and caused reactivity excursions.[3,4] After some experience with uranium separation and the resulting power disturbances, operating restrictions were established with the object of limiting the potential reactivity increase associated with the separated uranium. It became a requirement that the reactor power be lowered to permit uranium recovery if there were indications that the specified limit on separated uranium had been reached. The limit was the amount that would result in a reactivity change of 10% $\Delta k/k$ if it were to move to the center of the core. If this amount of reactivity were added in 3 sec (a time set by solution velocities in the core), a pressure surge of 1000 psi would be expected. The reactor average temperature was used as the immediate indication of uranium separation, since early experience had shown that the uranium accumulated at a position of relatively low importance and resulted in a noticeable decrease in temperature. It was recognized that the temperature was not an absolutely reliable indicator, because, if the

deposit were to occur on some other parts of the core tank, a hazardous situation could develop without any change in temperature. Therefore other indications, such as small unexplained power disturbances and shifts in the blanket-to-core power ratio, were also cause for power reduction when the reactor was operating under conditions that were either unexplored or where uranium separation was possible. The melting of titanium corrosion specimens and the holes in the Zircaloy core tank were consequences of the uranium separation. There is evidence that any metal-water reactions which may have started were quenched as soon as the fission heat source was dispersed.[3]

Primary Containment System

The first of the several lines of defense against release of radioactivity from a fluid-fueled reactor is the piping, which in many ways is equivalent to the fuel cladding in a solid-fuel reactor. The integrity of the primary containment piping of the HRE-2 was tested routinely at intervals of about 2500 hr of operating time by hydrostatic tests with distilled heavy water at 1.5 times the operating pressure. The maximum permissible leakage from the high-pressure piping system was originally specified as 5 cm^3 per day, and in more than three years of operation the leakage was normally much less than this amount. Escape of radioactivity from the first barrier could be detected by the flange leak-detection system, by changes in solution inventory, and by the activity monitors in the cell and on service lines that passed through the shield.

Flange leak detectors were provided by connecting each groove of the 182 ring-joint flanges in the reactor process piping to a header filled

with heavy water at a pressure 500 psi above that inside the piping. Because the header was kept at a pressure above system pressure, the leak detection feature of the flanges was also a buffer that helped to ensure that solution could not leak by the gasket. Flanges and stainless steel ring gaskets were matched to close tolerances to provide a double seal from the groove to the outside and to the inside. Leakage of heavy water from the grooves was detected by a reduction in header pressure, and makeup was supplied to keep the system pressurized. During operation, makeup amounted to only 3 cm^3/day. There was not a single instance of leakage from the piping at a flange. This record is regarded as evidence that satisfactory performance can be attained with metal O-ring-type flanged joints in a highly contaminated system.

The reactor piping was contained within a cell, as described below. Leakage of activity from the piping into the cell could be detected by the continuous circulation of the cell atmosphere through shielded radiation monitors. The sensitivity of the monitors was such that leakage of more than a few milliliters of fuel solution per minute should have been detectable. Only twice did detectable fuel leakage occur. On one occasion a crack in a weld on a diaphragm type of fuel feed pump released a few liters of fuel solution into the cell over a 25-hr period before the reactor was shut down for replacement of the pump. Later a crack developed in a tee welded into the drain line from the fuel dump tanks and resulted in the loss of several liters of fuel solution. In both instances the cell air monitors warned of the leak.

The total quantity of liquid in the reactor was measured several times daily as a means of detecting gross leakage. The only leak large

enough to be detected by this method occurred during a reactor cooldown, when a crack developed at a highly stressed point in a 1/4-in. pipe carrying heavy water. The activity in the cell was so low that it was not detected by the cell-atmosphere monitors, and the leak was finally indicated by the inventory.

Many of the primary fuel-containment pipes were jacketed and were cooled or heated by fluids from outside. All the service lines carrying fluids through the reactor shield passed monitoring stations so that excessive activity indicative of fuel leakage would cause an alarm and initiate a reactor shutdown. On two occasions this system detected the failure of feed-pump diaphragms and the resultant introduction of activity into the water in the drive lines penetrating the shield. The only "scram," or emergency, shutdown in the history of the reactor was caused when one of the monitors was exposed to radiation from a source other than the line it was monitoring.

Secondary Containment System

Because radioactivity could escape from the primary containment piping in spite of all precautions, a second barrier with stringent leakage specifications was provided. This was a welded steel box with a volume of 0.75×10^6 liters that contained all the reactor equipment. The box was actually a liner inside the biological shield, and it was never entered during operation. Only during maintenance periods, after the fuel had been drained and the piping rinsed, was any breach made in this container. Thus airlocks or quick-opening closures were eliminated, and the container was seal-welded to reduce leakage to a minimum.

The structure supporting the container was designed to be strong enough to withstand the simultaneous release of all the fluids contained in the piping and tanks within the cell. The calculated pressure rise within the container in such an event, assuming the worst conditions, was 32 psig. When the container construction was completed, the adequacy of the design was demonstrated by a hydrostatic test, using strain gages at critical points to indicate stresses in the structure. The container proved to be adequate for the predicted maximum pressure, but to provide a greater margin of safety, an automatic cold-water spray system was installed in the cell to help condense vapors and to reduce the pressure in the event of a catastrophe.

The permissible leakage rate from the secondary container was specified to permit building occupants to escape in a reasonable time without excessive ingestion of activity in the event of the complete failure of the piping within the cell. The concentration of fission products in the cell atmosphere that would be leaking into the building was estimated in the following manner. A decontamination factor of 20 was assumed for iodine, because most of the iodine in the reactor was held in a stable form on an iodine trap installed to help reduce xenon poisoning. For the other fission products, it was assumed that one-third of the fuel would evaporate and that particle impaction and condensation in the cool cell would result in a decontamination factor of 100. Allowing for a breathing rate of 30 liters/min, a building evacuation time of 1.5 min, and a maximum permissible exposure of 25 rem, the maximum permissible leakage rate from the cell was set at 20 liters/min at an internal pressure of 30 psig.

The leakage rate from such a large container is very difficult to measure accurately at pressure, so several tests were devised for the measurement. First, the vessel was pressurized with gas, and the rate of pressure decrease was determined. The pressure test indicated a total leakage of several hundred liters per minute, which was excessive. The top of the container was then flooded with water to reproduce the actual operating conditions, and a leakage rate of only a few liters per minute from the water surface was found. Next, the many service-line penetrations through the container wall were examined, and several valves were found to be leaking. These and leaks through electrical conduits accounted for all except 20 to 25 liters/min of the total leakage at 30 psig and were eliminated by repairs.

When the 20- to 25-liter/min loss could not be located with air pressure, helium gas was added and all known possible escape points were probed with leak detectors. Still the leakage source could not be detected, and it was concluded that the leakage was from the underground portion of the container. Even though leakage which does not escape into an occupied area is not particularly hazardous to personnel, another effort was made to determine the path of escape by pressurizing the shield to 15 psig and vaporizing 2.2 liters of ethyl mercaptan solution to produce a concentration of the odoriferant about 2000 times the detectable level. Operators were stationed throughout the reactor area, but only one significant leak was sensed; it appeared certain that the large fraction of the leakage was not entering the inside of the building and that the leakage would not be dangerous at an outdoor escape point. After stopping all known leaks, final measurements of the shield leakage made over long periods

with the shield temperature held constant showed rates of 2.5 and 4.3 liters/min for shield pressures of 7.5 and 15 psig.

Each time the container was closed after having been opened for reactor repairs, it was rewelded and retested. The reactor was shut down eleven times for repairs, and in every case it was possible to reseal the container without exceeding the allowable leakage rate. Futhermore, in every case it was possible to reduce the leakage at the most important surface (the top) to a few hundred cubic centimeters per minute.

During normal operation it was required that the container be maintained at an absolute pressure of 1/2 atm as an additional protection. This made it possible to continuously measure the rate of leakage by the pressure rise. In the three and one-half year history of the reactor the measured leakage rate never exceeded the allowable rate.

A final precaution against the effects of a possible rupture of the container was to provide a nitrogen atmosphere within the cell. By keeping the oxygen content of the cell atmosphere in the 5 to 8% range, the danger of explosive mixtures from D_2-O_2 escape or oil leakage was virtually eliminated. A nitrogen purge of less than 5 liters/min was sufficient.

There were a number of routine operations in which lines leading into the reactor system were opened, and radioactive fluids were intentionally withdrawn. These operations included sampling, which was done on a daily basis, and the infrequent addition of fuel solution or removal of wastes. It was recognized that intentional breaching of both the primary and secondary containment barriers was a hazardous operation, and special provisions were made to prevent activity releases during such

operations. Special equipment was used, and the operations were strictly controlled through procedures and supervision.

Sampling the fuel solution was accomplished by circulating solution through an isolation chamber, which could be drained into a sample bottle through a hypodermic needle that penetrated the rubber diaphragm seal on the bottle. This apparatus was contained in a cavity that was maintained at a slight vacuum by stack draft. On one occasion, early in the reactor operation, several milliliters of fuel solution spilled in the cavity when the needle transfer arrangement failed. Between 10 and 20 curies of activity was discharged from the stack, and this was the largest release of activity in the history of the HRE-2. As a result of this minor incident, additional protection was provided by installing an activated carbon, silver mesh, absolute filter in the exhaust line to the stack to trap activity released in the sampler cavities.

On two occasions during addition of gas into the reactor piping, radioactive liquid backed up through the gas addition lines, leaked through temporary connections outside the shield, and contaminated small areas of the floor. Back-flow preventers and more sensitive radiation detectors were installed to help prevent recurrence of such incidents.

Ventilation was installed as a "third-line defense" in the reactor operating areas where radioactive liquids might escape the other two barriers through normal handling or sampling operations. These areas were maintained at a slightly negative pressure (approximately 0.1 in. H_2O) by exhausting air to the area stack through carbon filters and absolute filters.

This system of ventilation was needed during normal power operation. During most shutdowns, another need arose as a result of maintenance

operations. When shielding blocks were removed from the top of the reactor and the containment barrier was cut away for maintenance, the stack fan drew air from the container and caused a draft through the opening in the shield. The draft protected against the escape of radioactive dust particles or gases from inside the container. When contaminated articles were lifted from inside the shield into the occupied area, they were either bagged in plastic or provided with a rigid airtight cover. These provisions were adopted after early maintenance operations resulted in the release of small amounts of activity.

Operating Procedures and Organization

As described above, the safety of the reactor operation was ensured, insofar as possible, by the design and characteristics of the physical system and by restrictions in the form of interlocks. As a supplement to the safety features built into the reactor, great emphasis was placed on the use of strictly enforced operating procedures and routine checks to further ensure safe operation, to prevent release of radioactivity, and to detect abnormal situations before they became hazardous.

Detailed operating procedures were prepared during the period of reactor construction and initial training of the operating staff. As the mode of reactor operation changed, successive revisions of these basic procedures were necessary. All revisions were incorporated in an up-to-date master copy of basic procedures that was kept in the reactor control room, along with specific instructions for the current run. Revisions were approved by the Operations Chief, and all operators were notified of changes and were required to become familiar with them.

The operating staff was composed of four groups serving on a rotating-shift basis. Each group consisted of one engineer, as Shift Supervisor, a senior technician, as Control Room Supervisor, and three operators. Each operator was given a specific assignment for each shift, and the assignments were rotated so that all the operators were experienced in all jobs.

Several management policies were emphasized in the continuous effort to avoid errors and to prevent the occurrence of dangerous situations. As instances, the reactor control room was manned at all times, even during maintenance shutdowns, and the activities on each shift were recorded in considereable detail in a control room logbook. Operators patrolled the entire area regularly, using a data log to record all variables of interest. Other personnel within the reactor operations organization assisted the operators in special experiments and some operations, such as sampling.

Written, approved procedures and checklists were used for every operation in which a connection was made from outside the secondary containment cell to the reactor piping. Two operators were assigned, each to check the action of the other. A supervisor or health physics representative was also required to be present to oversee the operation.

Checklists were also used to ensure that the operators made a detailed survey of all reactor systems before a reactor startup. During operation, many of the instruments and protective interlocks were tested routinely by instrument mechanics and operators.

Maintenance activities were also carefully controlled to ensure that functions of equipment or instruments were not modified and that proper

provisions were made for potential hazards. All maintenance work was authorized by work orders issued by the Maintenance Supervisor. In addition, a work permit that indicated the precautions to be taken had to be signed by the Shift Supervisor on duty before work could begin on any reactor equipment or instruments. When changes were made in any equipment or instrumentation, an approved drawing had to accompany the permit. Any modifications that affected reactor safety, such as changes in any control circuits, had to be approved by a standing ORNL review committee.

(P. N. Haubenreich, S. E. Beall, J. W. Hill)

SL-1 FINAL REPORT

The final report on the Stationary Low-Power Reactor (SL-1) accident was issued by the AEC's Board of Investigation on September 5, 1962.[7] The accident, which occurred at the AEC's National Reactor Testing Station (NRTS) in Idaho on January 3, 1961, and resulted in the first operational reactor fatalities in the United States, was discussed in an earlier issue of Nuclear Safety.[8] While additional information has become available, the basic conclusions as to the cause of the accident are unchanged. The direct cause of the incident clearly appears to have been manual withdrawal of the central control-rod blade beyond the specified limit by one or more members of the three-man maintenance crew. The actual reason for the abnormal withdrawal could not be established because of insufficient evidence.

It appears highly unlikely that any significant additional information concerning the incident will become available, since, on July 27, 1962, the third and final phase of the SL-1 recovery operation was completed. To summarize briefly, the accomplishments during the three phases of recovery were the following:

Phase 1. The three casualties were recovered from the reactor building, and the reactor was determined to be in a noncritical condition. This recovery phase was completed one week after the incident.

Phase 2. Combustion Engineering, Inc., the reactor operator, determined that the reactor pressure vessel contained no water and that nuclear excursions could be prevented by keeping it that way. This recovery phase was completed in May 1961.

Phase 3. The General Electric Company was assigned responsibility in May 1961 for the balance of the recovery operations. The General Electric group (1) gathered and evaluated data concerning the accident, (2) dismantled, examined, and disposed of the SL-1 pressure vessel and core, and (3) razed the reactor building, decontaminated the SL-1 area, and made the area suitable for future projects. These recovery operations were completed in July 1962, and the final report has been issued.[9]

Phase-3 Recovery Operations

The information gained in dismantling and inspecting the pressure vessel and core was among the most valuable obtained during the recovery operations. It was determined, beyond all reasonable doubt, that the central control rod was tightly bound in its shroud in a position corresponding to 20 in. withdrawal. There was no evidence of relative motion after seizure. Available data indicated that the reactor would have been critical with a rod withdrawal of 16.7 in. The 20-in. position corresponded to a reactivity of $(2.4 \pm 0.3)\% \Delta k/k$. It was estimated that because of the rod withdrawal, the power rose on a period of about 4 msec until the excursion ended.

During the excursion the fuel plates in the central region of the core were destroyed. A schematic diagram of the approximate range of damage is shown in Fig. VI-1. During the recovery operations, a critical experiment with the damaged core revealed that it was highly subcritical.

The fatal excursion was probably ended by destruction of the core and removal of the moderator, which was ejected from the vessel by steam pressure. Significantly, the normal and emergency shutdown systems would

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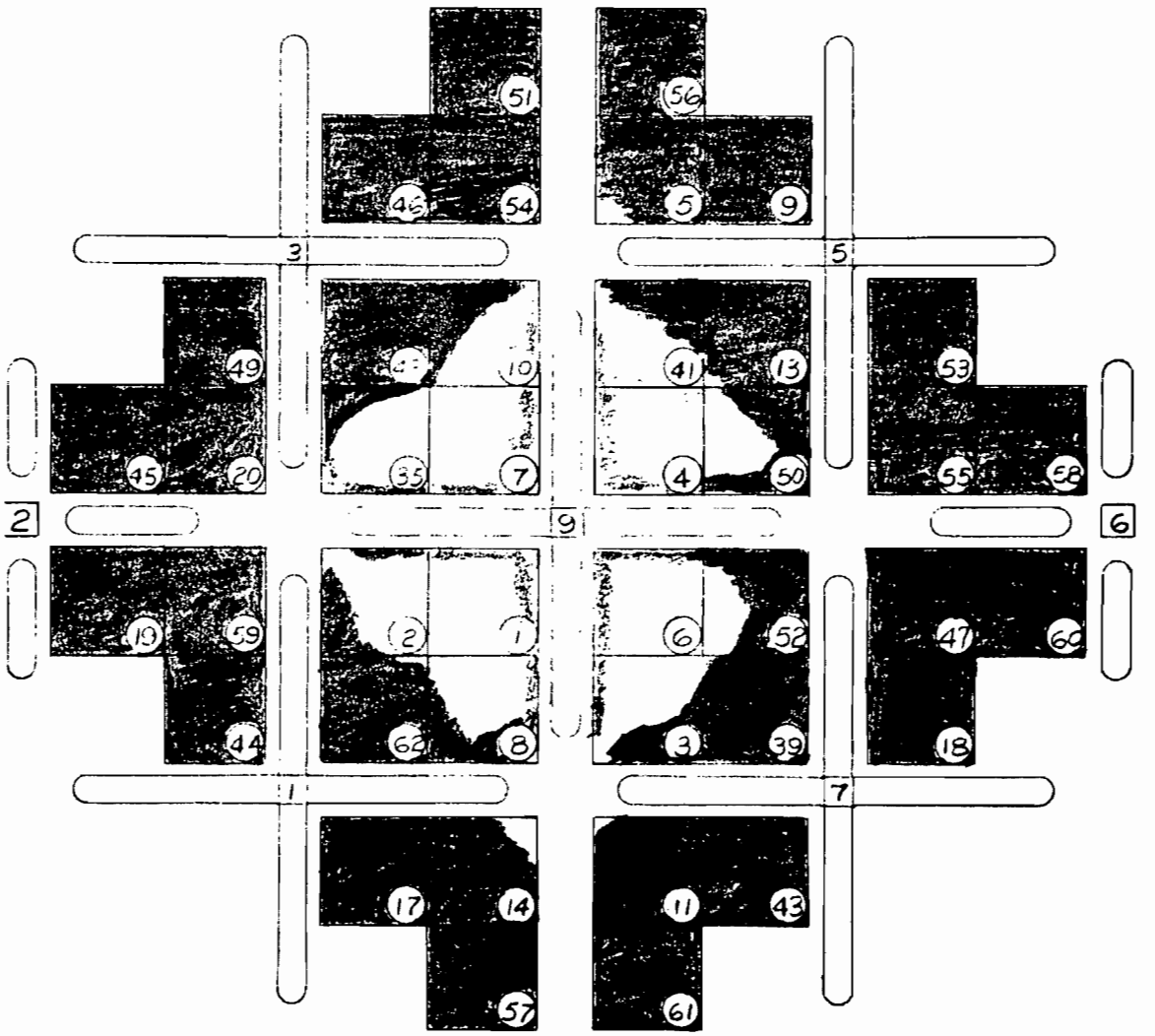

Darkened areas - Unaffected plates

Light areas - Destroyed plates

Fig. VI-1. Schematic View of Core, Showing Areas of Destruction. The dotted lines in the central region and the number 9 indicate the normal position of the central control rod. (Taken from ref. 9).

have been unavailable if they had been needed to end the excursion. The safety and control-rod system was ruined, and the emergency boron-injection system and the emergency cooling system had been rendered ineffective.

It was estimated that the total nuclear energy generated during the incident was 130 ± 10 Mw-sec. This relatively low yield produced some striking results, however. Rapid formation of 500-psi steam in the core accelerated the water above the core and produced a water hammer that hit the pressure vessel lid with pressures up to 10,000 psi. The vessel sheared its connecting piping and was lifted approximately 9 ft into the air by the momentum transferred from the water. The estimated time from the control-rod withdrawal until the vessel fell back into its support cylinder was estimated to be from 2 to 4 sec. All the mechanical displacements observed were produced by kinetic energy, which was less than 1% of the total energy released in the incident.

There was definite evidence of reactions between hot aluminum and water during the incident. Alpha-phase alumina, which could have come only from such a reaction, was found in the vessel. It was estimated that this metal-water reaction added 24 Mw-sec of energy to the incident.

Statements to the Board of Investigation

During the final sessions of the Board of Investigation, in August 1962, statements were made by Combustion Engineering, Inc., the reactor operator, and by Argonne National Laboratory, the reactor designer.[10]

Combustion Engineering personnel noted that the recovery examination by General Electric did not bring to light any new basic cause for

the incident. With respect to the presumed manual movement of the central control rod, it was felt that it was possible only to speculate as to the reason for the excessive rod withdrawal. This particular rod had no history of ever being stuck or jammed in its shroud. Also, the rod examination did not indicate any tears or gouges such as there might have been if the rod had been stuck and then pulled loose.

After discussing many assumed reasons for an operator lifting the control rod 20 in., Combustion personnel concluded:

> "We have not been able to visualize any situation acceptable on technical grounds which would explain why an operator deliberately would pull the rod out of the reactor. It can be argued that some fault was noticed in the rod and that removal was initiated in order to examine this fault. This, of course, would have required a disregard of all instructions and experience which indicated that removing the central rod would cause the reactor to become critical. Nevertheless, it can be supposed that such action was taken. What is very difficult to understand is that in order to produce the excursion such removal would have to be done quickly, which is not consistent with a deliberate intention to examine the rod.
>
> "A deliberate withdrawal of the rod without regard for the hazard known to be connected with such an act raises the question of irresponsibility. The training of the operators, the fact that two of them had been fully qualified by examination, and their previous satisfactory performance as operators seem to make an irresponsible act by one of them unlikely unless the sudden appearance of emotional instability is postulated. The examination of SL-1 [facility] did not shed any light on this question. As was mentioned earlier, in the absence of eyewitnesses what actually happened may remain a mystery forever."

Since the reactor shutdown margin before the accident was between 2 and 3%, Combustion personnel felt that any supposed gain in reactivity from a loss of the boron-aluminum burnup strips from the fuel elements could not have contributed to the cause or magnitude of the incident. It was also pointed out that the SL-1 operators had carefully and consistently monitored the position of the controls, since this was a

sensitive way of determining nuclear reactivity. Reconstruction of the accident reactivity situation was consistent with the indications of the monitoring program.

If the withdrawal of more than one control rod had been necessary to achieve criticality, it can be assumed that the reactor incident could not have happened. Combustion personnel made the point that this was recognized and that steps had been taken prior to the accident to correct the situation. The design and construction of a new core had been recommended by Combustion Engineering and approved by the AEC. The reactor revision would have provided for adequate shutdown with one rod withdrawn. The boron-aluminum fuel element strips would have been replaced with boron-steel encased in stainless steel; newly designed control rod drive mechanisms would have been installed; and stainless steel would have replaced aluminum as a structural material.

Combustion personnel concluded:

> "In view of the approval of SL-1 for operation by the hazards evaluation authorities, the successful operation for two and one-half years, and the program of careful monitoring of reactivity of the reactor, we did not consider continued operation of this experimental power reactor to be any more hazardous than it was at the initiation of the contract. In view of the fact that the disassembly and analysis of the reactor has not indicated any deficiency in the operation which could have contributed to the accident, we believe that our judgement in maintaining the reactor in operation, which was concurred in by the operating office of the Commission, was not faulty and did not show lack of regard for the requirements of proper reactor operation."

The Argonne National Laboratory's statement supplied a summary of the original reactor design criteria and goals and the evolution of some aspects of the design. (See article on SL-1 control system in this issue under heading "Some Aspects of the SL-1 Control-System Design.")

The SL-1 was designed as a prototype of a power reactor that could be located in remote arctic areas and operated by military personnel. It was to contain essentially only those safety features required to protect the plant and its operating personnel. The reactor was to have a core life of at least two years at design load. All moving components of the reactor were to be readily available for inspection. Where feasible, standard, commercially available components were to be used. These and other criteria were directed at providing a simple, reliable plant.

Argonne personnel made the following statements concerning the control rod and fuel element configurations:

> "A special feature of the reactor design was that refueling could take place without removing the head of the pressure vessel. This was accomplished by sizing the control-rod-drive nozzles, in the vessel head, to permit the loading of fuel assemblies through the nozzles. It was felt that this concept of fuel handling would simplify significantly changes in the core loading in the ultimate power plants.
>
> "A number of layout studies were made in order to size fuel element dimensions and control rod strengths and locations. The requirements of convenience of fuel handling were not found to be compatible with rod arrangements under the criteria that the rods by themselves could provide full operating-to-shutdown margin, while simultaneously no single rod could start up the reactor. The decision was therefore made to sacrifice this latter criterion."

To compensate for the excess reactivity in a new fuel batch, the reactor design included B^{10} as a neutron absorber. Originally the boron was to have been included as a constituent of the fuel; however, fuel plates produced by two individual companies on development contracts were unacceptable. Therefore:

> "In the interest of expediting the procurement of satisfactory fuel plates from a commercial fabricator, the specifications were modified to eliminate the burnable poison from

the plate. Instead, it was decided to use thin strips, made by extruding an aggregate of powders of (X-8001) Al-Ni alloy and highly enriched boron. This process was similar to the process adopted to produce poison strips used successfully in BORAX III. It was realized that strips like these had not been exposed to corrosion for the length of time expected in ALPR [SL-1], nor had they been irradiated to the nvt level (time-integrated thermal neutron flux) anticipated. In short, the thin burnable-poison strips were used in the expectation that they would be satisfactory, but without an opportunity for a conclusive, confirming test."

Commenting on the control-rod drive, Argonne personnel pointed out that the unit was selected for two major reasons. First, it was relatively simple in concept and design, and, second, the Army Package Power Reactor had used a similar drive and it had therefore been tested in an analogous environment.

Conclusions of the Board

The AEC Board of Investigation, after consideration of information gathered since the SL-1 accident, came to the following conclusions:

"(1) The immediate cause of the incident was withdrawal of the central control blade. The displacement of the blade was consistent with the energy release determined from post-incident studies.

"(2) The absence of any evidence of an internal agent, such as a chemical reaction or explosion, together with the fact that only the central rod (and none of the four peripheral rods) was determined to be in a withdrawn position at the time of the explosion, makes it reasonably certain that the blade was withdrawn by an external agent.

"(3) The only plausible hypothesis is that the central control blade was withdrawn manually, during the final phases of a maintenance operation of reassembly of the drive mechanism.

"(4) No direct evidence is available as to how or why the central control blade was withdrawn to the position determined to exist at the time of the explosion. In particular, there is no direct evidence to indicate any kind of operational difficulty or other reasons to deviate from the approved procedure for the reassembly operation in question."

In concluding their assignment the Board made the following remarks:

> "We continue to hold that the immediate responsibility for safe operation of the SL-1 was that of the on-site operating contractor, Combustion Engineering, Inc., and that, by virtue of the effect on contractor operations of financial, program and other direction by the AEC Idaho Operations Office and the Headquarters Division of Reactor Development, these organizations shared in the responsibility for the safe operation of the SL-1 plant.
>
> "We strongly recommend that the Commission take the necessary steps to assure that the facts, conclusions and implications relating to the SL-1 incident be disseminated to the interested public. While we believe that the Board has completed its assignment with the submission of the attached report, individual Board members and consultants will, of course, be available to consult with and advise those persons who may be selected by the Commission to effect such dissemination."

After receiving their final report, the AEC General Manager dissolved the SL-1 Board of Investigation. Although this essentially closed the case, the AEC plans to prepare a report on lessons learned from the accident so that the nuclear community can gain experience from it. Nuclear Safety will review the publication when it becomes available. (J. R. Buchanan)

HYDROGEN FIRE AT PM-3A NUCLEAR POWER PLANT

On October 7, 1962, a flash hydrogen fire occurred inside the containment tanks of the PM-3A Nuclear Power Plant at McMurdo Sound, Antarctica.[11] There was no injury to personnel, no damage to major equipment, and no abnormal radioactivity produced. An analysis made after the accident showed that the fire was caused by the combustion of hydrogen that was probably ignited by a line-to-line short.

The PM-3A is a portable, medium-power, pressurized-water reactor with a capability of a net electrical output of up to 1500 kw.[12] It was designed, built, and is being initially operated by the Martin Company. The U.S. Navy will eventually assume responsibility for its operation. The reactor was designed so that the components could be readily transported and then assembled with only those construction tools available at a remote site. The plant consists of a single primary coolant loop, a steam generator, and a turbine-generator. The major components of the plant are shown in Fig. VI-2.

The plant containment system consists of four interconnected vessels that house the primary system components. The four vessels serve as shipping containers for the major reactor components when they are being moved to the plant site. When assembled the containers are partially buried in the ground in an upright position on a common base. The diameters of the vessels range from 6.5 to 8.0 ft and the heights from 27.3 to 30.8 ft. The reactor vessel is housed in one of the containment vessels; the steam generator, coolant pump, and pressurizer are in a second; spent-core storage space is provided by a third; while the fourth,

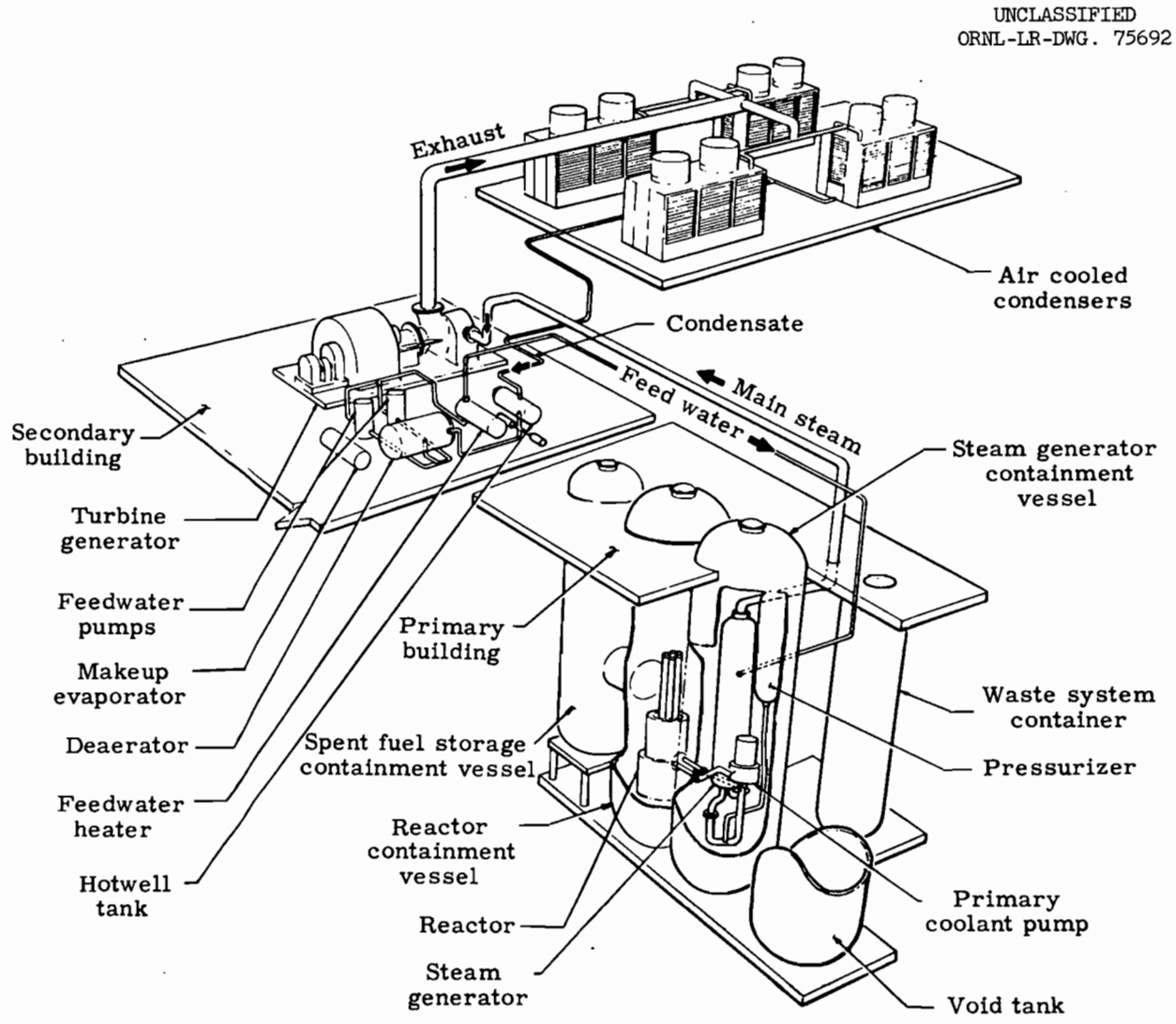

Fig. VI-2. Basic Flow and Orientation Diagram of PM-3A Nuclear Power Plant. (Taken from ref. 12.)

a void vessel, provides additional expansion space to reduce the peak pressure in case of an incident. Normally the reactor and spent-core containers are filled with water to a high level (for shielding), and the other two vessels are dry.

The PM-3A was made critical on March 3, 1962, and low-power operation was started on July 7, 1962. From July 7 until the accident, the reactor was on the line continuously, except for a few brief shutdowns.

Events on Day of Incident

On October 7, 1962, at 0615 hr, the PM-3A experienced an automatic scram, the cause of which has not been determined. With plant conditions appearing normal, an attempt was made to start the reactor about 1 hr later. A scram occurred, however, because of a spurious signal from one of the nuclear instrumentation channels. Similar behavior had been experienced previously when ion chambers became damp, and since no difficulty could be located, another startup was attempted about 30 min later; the results were the same as before.

More checking was done, and then an additional startup attempt was made at 0810 hr. A scram in this instance was caused by a trip of one of the three a-c power circuits. A short circuit in the steam generator level-indication system caused an overcurrent that tripped the circuit. Unsuccessful attempts were then made to restore the generator level indication. It was not known until later that the grounded circuit was inside the containment tank.

A fourth attempt was made to achieve criticality near noon. Indications were that the ion chambers were becoming increasingly damp, and

it was feared that this would prevent a startup. Since it was desired to hold the reactor above the heat-up range and check the nuclear instrumentation behavior, pulling of the control rods was initiated at 1244 hr. The plant was shut down after it was noted that the rod position indication was erratic. It was then determined that the difficulty was inside the containment tanks.

After about 8 hr, the reactor equipment had cooled sufficiently to allow personnel entry to all containment tanks. The examination revealed that a fire of short duration had occurred that was centered in the upper portion of the steam-generator containment tank. Damage was largely superficial. The fire had scorched cables and paint, and the pressure of the shock wave had damaged junction box covers, sheet metal work, and some thermal insulation. The control-rod actuator-coil cans were partially collapsed around the holding coils and position-indicating mechanism.

Cause of Incident

A thorough investigation was initiated to determine the cause of the fire. All possible combustible material that could conceivably be present in the containment tanks was investigated to determine which material had been present in sufficient quantity to be flammable. The investigation revealed that the only combustible material that could have been present in sufficient quantity to support the fire was hydrogen gas. A combustible-gas detector was located in the containment tank near where the fire initiated, but it was not operating on the day of the accident, because of an operator error.

There were two major sources of hydrogen. One was radiolytic decomposition of the shield water normally in the reactor and spent-core containment tanks. Calculations indicated that the hydrogen generated from this source after 36 days of operation would yield a concentration of 2% if uniformly distributed in all containment tanks. The other source was leakage from the vapor space of the coolant system. The gas in the vapor space is formed by radiolytic decomposition of the coolant and from decomposition of the hydrazine added to the coolant as an oxygen scavenger. Most of the hydrogen in the coolant loop would normally accumulate in the pressurizer steam dome. On the day of the accident the stem packing of a block valve on the upper portion of the pressurizer was allowing significant leakage of steam and hydrogen. Calculations indicated that this leakage was sufficient to increase the hydrogen concentration in the upper portion of the steam generator container to the explosive limit. It is felt that the gas was probably ignited at 0810 hr when a line-to-line short occurred in the steam-generator level-indication system.

The peak pressures that resulted from the fire were estimated from the physical damage. At a distance of 5 ft from the center of the explosion the peak was greater than 5 psig, and at 20 ft from the center it was less than 60 psig. The containers were designed for a peak pressure of 126 psig.

Corrective Action

Proceeding on the assumption that all potential ignition sources could not be eliminated from the containment tanks, action was taken to

prevent the accumulation of explosive mixtures. Among the actions taken were the following:

1. Fans and ducts were installed to eliminate stagnant pockets where hydrogen could accumulate, such as the pocket above the pressurizer in the steam-generator containment tank.

2. A second hydrogen detector was added.

3. Modifications were made so that the air in all containment tanks, with the exception of the void tanks, could be sampled.

4. It was established that prior to a reactor startup an inspection would be made to ensure that the pressurizer system leakage was sufficiently low to prevent loss of significant quantities of hydrogen.

5. Changes were made in the operating procedures to ensure that the operator would be aware at all times of the approximate hydrogen distribution in the system and containment tanks. In the future, if the hydrogen concentration of the containment air rises to approximately 2%, the plant will be shut down. Failure of both of the hydrogen detectors will also be a reason for shutdown.

Martin Company personnel believe that these changes "will result in safe plant operation. However, in the interest of reliability and long life, some additional modifications" are being studied. These include alterations to the containment tank air-circulating system for increased reliability and the addition of a hydrogen-removal system.

In addition to these actions taken at the PM-3A site, it is anticipated that the AEC's Division of Licensing and Regulation will survey all existing reactors that have potential containment vessel hydrogen

sources. The survey will seek to determine whether these plants have sufficient protection against hydrogen explosions. (J. R. Buchanan)

CHANGES IN LICENSING REGULATIONS

Amendments to the AEC rules and regulations published in the Code of Federal Regulations, Title 10, Chapter 1, are frequently proposed. Among those pertinent to nuclear safety that are being considered for change are Part 2, Rules of Practice; Part 20, Standards for Protection Against Radiation; Part 30, Licensing of By-Product Material; Part 40, Licensing of Source Material; Part 50, Licensing of Production and Utilization Facilities; Part 115, Procedures for Review of Certain Nuclear Reactors Exempted from Licensing Requirements; and Part 150, Transfer of Regulatory Power. The proposed changes in Parts 2, 50, and 115 were designed to implement Public Law 87-615 passed by the 87th Congress. The various changes are discussed below.

Part 2, Rules of Practice

> "Public Law 87-615 of the 87th Congress, which became law on August 29, 1962, adopted certain amendments of the Atomic Energy Act of 1954 designed to permit greater flexibility and to encourage informality in the conduct of licensing proceedings conducted by the AEC [ref. 13].
>
> "Section 1 of Public Law 87-615 authorized the Commission to establish one or more atomic safety and licensing boards, each composed of three members, two of whom shall be technically qualified and one of whom shall be qualified in the conduct of administrative proceedings, to conduct such hearings as the Commission may direct and make such intermediate or final decisions as it may authorize in proceedings with respect to granting, suspending, revoking or amending licenses or authorizations. The amendment also authorizes the Commission to delegate to such a board such other regulatory functions as the Commission deems appropriate.
>
> "Section 2 of Public Law 87-615 amended subsection 189a. of the Atomic Energy Act, which previously required a public hearing prior to granting a construction permit as well as prior to granting an operating license for a power or testing

reactor, by limiting the mandatory hearing requirement to the construction permit stage. The amended subsection 189a now authorizes the Commission also to dispense with advance public notice with respect to an amendment of a construction permit or an operating license on determining that the amendment involves no significant hazards considerations, and in other cases to issue an operating license or amendment to a construction permit or operating license without a hearing but on public notice of its intent to do so.

"Section 3 of the Act amends subsection 182b. of the Atomic Energy Act by clarifying the scope of review by the Advisory Committee on Reactor Safeguards (ACRS) of applications for licenses for power and testing reactors, by providing that the referral to the Advisory Committee on Reactor Safeguards of applications for amendments of either a construction permit or an operating license is discretionary on the part of the Commission."

Amendments "designed to effectuate the provisions...of Public Law 87-615" have been proposed. "A new Section 2.721 would be added to [10 CFR] Part 2, providing for the establishment of atomic safety and licensing boards and for their duties and powers."

"A new Section 2.756 would be added to Part 2 encouraging the use of informal procedures consistent with the Act, the Administrative Procedure Act, and the Commission's regulations, the orderly conduct of the proceedings, and the necessity of preserving a suitable record for review."

Part 20, Standards for Protection Against Radiation

On October 17, 1962, a proposed amendment to Part 20 on the modification of labeling requirements was published in the Federal Register. A period of 60 days was allowed for comment.[14]

Part 30, Licensing of By-Product Material

Amendments to Part 30 and Part 40 were published in the Federal Register of August 16, 1962, and went into effect immediately.[15] Both amendments were designed to prohibit the export of by-product and source

material from the United States if the exporter "knows or has reason to believe that any part of it is to be re-exported from the original country" of destination. The action was taken to prevent unauthorized shipment to Soviet bloc countries.

The AEC staff still has under consideration an amendment proposed in January 1962 providing for the exemption or general licensing of certain by-product materials for medical use.[16,17] There had been considerable opposition to the amendment from those in the medical profession. Those opposing the change felt that the average medical practitioner lacked the necessary training in the application of radioisotopes.

Part 40, Licensing of Source Material

An amendment to Part 40 was published in the Federal Register of August 16, 1962.[15] The purpose is discussed under Part 30 above.

Part 50, Licensing of Production and Utilization Facilities

The pertinent provisions of Part 50 are to be amended to limit mandatory referral of applications to ACRS in accordance with the purposes of Public Law 87-615.[13] The mandatory hearing requirement of Section 50.59 would be eliminated. The purposes of the new law are discussed under Part 2 above.

Part 115, Procedures for Review of Certain Nuclear Reactors Exempted from Licensing Requirements

The pertinent provisions of Part 115 will be amended to limit mandatory referral of applications to ACRS in accordance with Public Law 87-615.[13] The purposes of the new law are discussed under Part 2 above.

Part 150, Transfer of Regulatory Authority

An agreement for the transfer of regulatory authority from the AEC to New York State was signed on October 15, 1962.[18] The agreement differed from those previously signed with other states in that it contained a Commission obligation to use its best efforts to cooperate with New York and other agreement states in the formulation of state and AEC programs for protection against radiation. The AEC also agreed to ensure that such state and AEC programs will be coordinated and compatible.

(J. R. Buchanan)

ACTION ON REACTOR PROJECTS BY LICENSING AND REGULATING BODIES

The Atomic Energy Act requires the AEC to protect the public from undue hazards resulting from exposure to radioactivity. Regulations that are followed to ensure that this responsibility is fulfilled, while furthering the simultaneous responsibility for developing the use of nuclear energy, are published in the Code of Federal Regulations. Title 10 of the Code requires the AEC to make certain specific findings regarding the safety of the public before issuing either the construction permit or the operating license for a facility. It also requires authorization for changes in facility equipment or operations that contain an element of hazard not previously reviewed or approved. The license-application record of the various facilities is reported in Table VI-1. Recent actions and activities relative to specific reactors are described below. Some information on reactors is also included that is not docketed legal action.

Big Rock Point Reactor (Docket 50-155)

The Big Rock Point boiling-water reactor of the Consumers Power Company achieved criticality September 28, 1962. According to the Consumers Power Company, it was the first reactor to be brought to initial criticality by utility company personnel.[19] The AEC issued[20] an 18-month provisional operating license for the reactor on August 30.

After some disagreement as to the description of "the location" of the plant operating areas for indemnification purposes, Consumers Power Company on September 19 agreed to the AEC's description. The

Table VI-1. CALENDAR OF LEGAL STEPS IN LICENSING U. S. POWER REACTORS

REACTOR INFORMATION					CONSTRUCTION PERMIT					OPERATING LICENSE				CURRENT STATUS
CODE NO.	NAME OWNER AND OPERATOR LOCATION	REACTOR TYPE	AUTHORIZED POWER Mw(t)	AUTHORIZED POWER Mw(e)	ACRS APPROVAL	HEARING EXAMINER APPROVAL	CONSTRUCTION PERMIT NO.	CONSTRUCTION PERMIT DATE	ACTUAL OR SCHEDULED COMPLETION DATE	ACRS APPROVAL	HEARING EXAMINER APPROVAL	AUTHORIZATION TO OPERATE NO.	AUTHORIZATION TO OPERATE DATE	
50-155	BIG ROCK POINT Consumers Power Co. Charlevoix County, Michigan	Boiling Water	157	47.8	Mar. 1960	May 1960	CPPR-9	5-31-60	1962	June 1962		DPR-6	8-30-62	Authorized to load first core and operate at 1 Mw(t) 9-26-62
	BODEGA BAY Pacific Gas & Electric Co. Bodega Bay, Calif.	Boiling Water		324					1965					Application not yet submitted to AEC
115-4	BONUS USAEC Puerto Rico Water Resources Authority Rincon, Puerto Rico	Boiling Water Nuclear Superheat	50	16.3	Mar. 1960	June 1960	6-28-60*		1963					Final hazards summary report filed in Feb. 1962
None	BORAX-V USAEC Argonne National Laboratory NRTS, Idaho	Experimental Boiling Water, Nuclear Superheat	35.7	2.7			Not Required					Not Required		Achieved criticality in Feb. 1962
50-144	CVTR Carolinas-Virginia Nuclear Power Associates Parr, South Carolina	Pressurized D_2O Moderator and Coolant	44.3		Dec. 1959	Apr. 1960	CPPR-7	5-4-60	Dec. 1962	July 1962				Plant to be ready for fuel loading in Dec. 1962
50-10	DRESDEN Commonwealth Edison Co. Grundy County, Illinois	Boiling Water	700	192 gr 180 net			CPPR-2	5-4-56	Sept. 1959	May 1959	Sept. 1959 Nov. 1959 Oct. 1960 May 1961 Sept. 1962	DPR-2	6-2-60 10-14-60 6-9-61 11-30-61 9-19-62	Initially critical Oct. 1959
115-1	ELK RIVER USAEC Elk River Rural Cooperative Power Association Elk River, Minnesota	Boiling Water Thorium Converter	58.2	22	Aug. 1959	Dec. 1959	12-18-59*		1963	May 1962				Planning startup
None	EBWR USAEC Argonne National Laboratory Argonne, Illinois	Experimental Boiling Water	100	4.5			Not Required					Not Required		Operating since Dec. 1956
None	EBR-II USAEC Argonne National Laboratory NRTS, Idaho	Experimental Breeder Fast Neutron Spectrum Sodium Cooled	62.5	16.5			Not Required					Not Required		Startup planned in 1962
None	EGCR USAEC Tennessee Valley Authority Oak Ridge, Tennessee	Experimental Graphite Moderated Helium Cooled	84.3	22.3	May 1959		Not Required		Dec. 1962			Not Required		Under construction
50-16	FERMI Power Reactor Development Co. Detroit Edison Co. Monroe County, Michigan	Fast Neutron Spectrum Plutonium Breeder Sodium Cooled	200	60			CPPR-4	5-26-59 11-18-60 7-14-61 11-29-61 8-6-62	Sept. 1962					Hearing on issuance of provisional operating license scheduled for late 1962
115-3	HALLAM USAEC Consumers Public Power District Hallam, Nebraska	Graphite Moderated Sodium Cooled	240	76	July 1959 Feb. 1960	June 1960	6-6-60*		July 1962	Feb. 1962	July 1962	DPRA-1	1-2-62 8-9-62	Provisional operating authorization permits "Wet Critical Experiments" and operation up to 15% of full power
50-133	HUMBOLDT BAY Pacific Gas & Electric Co. Buhne Point, California	Boiling Water	163	48.5	June 1960 July 1960	Oct. 1960	CPPR-10	11-9-60	Aug. 1962	Apr. 1962	Aug. 1962	DPR-7	8-28-62	Provisional operating license in effect; may not load until authorized
50-3	INDIAN POINT (CETR) Consolidated Edison Co. of New York Buchanan, New York	Pressurized Water Thorium Converter	585	275 gr 255 net		Final Design June 1961	CPPR-1	5-4-56 8-4-61	Apr. 1962	Nov. 1961	Feb. 1962	DPR-5	3-26-62	Initially critical Aug. 2, 1962
	LA CROSSE USAEC Dairyland Power Cooperative Genoa, Wisconsin	Boiling Water	165	50			Not required		1965			Not required		Application not yet submitted to AEC
None	N. S. SAVANNAH USAEC States Marine Lines Constructed at Camden, New Jersey	Pressurized Water	69	22,000 shp			Not Required		Dec. 1961	Dec. 1960	June 1961	Not Required		Criticality achieved in Dec. 1961; full power in Apr. 1962. To visit Seattle in Oct. 1962
50-130	PATHFINDER Northern States Power Co. Sioux Falls, South Dakota	Boiling Water Nuclear Superheat	203	66	Feb. 1960	Apr. 1960	CPPR-8	5-12-60	June 1963					Initial criticality tentatively scheduled for 1963
50-171	PEACH BOTTOM Philadelphia Electric Co. York County, Pennsylvania	High Temperature Unclad Fuel Graphite Moderated Gas Cooled	115	40	Nov. 1961	Feb. 1962	CPPR-12	2-23-62	Mar. to Dec. 1964					Under construction
115-2	PIQUA OMR USAEC (Reactor) City of Piqua Municipal Power Commission Piqua, Ohio	Organic Cooled and Moderated	45.5	11.4	July 1959	Jan. 1960	1-7-60*		Oct. 1961	May 1961	June 1962	DPRA-2	8-23-62	Operating authorization is provisional; core loading not yet authorized
None	PWR USAEC Duquesne Light Co. Shippingport, Pennsylvania	Pressurized Water	231	68 gr 60 net	Jan. 1954		Not Required		Nov. 1957	Nov. 1957		Not Required		Operating since Dec. 1957
50-146	SAXTON Saxton Nuclear Experimental Corporation Berks County, Pennsylvania	Pressurized Water	23.5	3.2	Sept. 1959	Jan. 1960	CPPR-6	2-11-60	1962	July 1961	Oct. 1961 Aug. 1962	DPR-4	11-15-61 10-3-62	License amended for conduct of Phase 1 of R&D program at power levels to 23.5 Mw
None	SRE USAEC Atomics International & Southern California Edison Santa Susana, California	Graphite Moderated Sodium Cooled	20	6			Not Required					Not Required		Operating since Apr. 1957
50-18	VBWR General Electric Co. Alameda County, California	Test Reactor Boiling Water	50	5			CPPR-3	5-14-56				DPR-1	8-31-57	Operating since Sept. 1957
50-183	VESR General Electric Co. Alameda County, California	Light Water Moderated, Separate Steam Superheater	12.5		May 1961	July 1961	CPPR-11	8-10-61	Apr. 1962 to Feb. 1963					Hearing on issuance of provisional operating license planned for early 1963
50-29	YANKEE (YAEC) Yankee Atomic Electric Co. Rowe, Massachusetts	Pressurized Water	485	141	Sept. 1957	Financial qualifications May 1959	CPPR-5	11-4-57 6-10-59	July 1960	Feb. 1960 May 1960 June 1960	April 1960 July 1960 Dec. 1960 June 1961	DPR-3	7-9-60 7-29-60 1-12-61 6-23-61	Operating since Aug. 1960. Hearing was held in Sept. 1962 to consider license amendment for operation at 540 Mw(t)

*Reactors classified as parallel procedure cases (code numbers 115-1 to 115-4) do not require construction permits, but their construction and operation are authorized by AEC.

power company had wished to include its gross land holdings, whereas the AEC wished to limit indemnification to those areas that normally will handle radioactive materials.[21]

On October 5 the power company requested[22] authority to expose individuals in restricted areas to airborne contamination in excess of the limits provided in Part 20, Appendix B, Table I of the Code of Federal Regulations. Supporting arguments were sent to AEC but were not available to the reviewer. The AEC has the request under consideration.

Dresden Reactor (Docket 50-10)

The license of the Commonwealth Edison Company was amended on September 4, 1962 to permit operation of the boiling-water Dresden reactor at 200 Mw(e), that is, 20 Mw(e) higher than previously.[23] The thermal power limit was increased from 630 to 700 Mw. Some necessary changes were also made in the technical specifications.

On August 23 the AEC advised the Company that its request for exemption from section 20.203 (c)(2) of Part 20 of the Code of Federal Regulations, which requires the equipping of high-radiation areas with visual and audible alarms, was granted.[24,25] The AEC was satisfied that the proposed alternate controls and procedures were adequate.

Commonwealth Edison notified the AEC in late August[20] that the Company no longer needed to request approval for alternate procedures for recording occupational exposures of Dresden employees because all employees have now signed form AEC-4. The Company's original request was made on February 9, 1962, as described previously.[25,26]

CVTR (Docket 50-144)

A public hearing was held on August 28, 1962 to consider issuance of an 18-month provisional operating license for full-power operation of the heavy-water-moderated 44.3-Mw(t) Carolinas-Virginia Nuclear Power Associates pressure tube reactor (CVTR).[27] A major point of consideration was the Southern Railway's concern about rail traffic control on their single-track main line that at its nearest point comes within 300 ft of the reactor. A further hearing was held on September 21, with respect to the railway matter. At the latter hearing the Southern Railway withdrew its opposition to the issuance of the reactor license.[28] The Southern stated that the potential hazards to rail traffic were shown to be less than they "had been given to understand" in meetings with power company representatives in 1961. On October 25, AEC Hearing Examiner McConnell recommended issuance of the provision operating license.[29]

Elk River Reactor (Docket 115-1)

On November 19, 1962 the 58-Mw(t) boiling-water Elk River reactor achieved a sustained nuclear chain reaction.[30] The AEC in mid-November had issued an 18-month provisional operating license for the reactor to the Allis-Chalmers Manufacturing Company.[31] The technical specifications of the reactor were submitted to the AEC in July 1962 and hearings concerning the reactor's proposed operation were held in late August and early September.[32,33]

Enrico Fermi Atomic Power Plant (Docket 50-16)

The ACRS on October 11, 1962 issued a favorable report concerning initial low power operation [up to 1 Mw(t)] of the Enrico Fermi fast-breeder reactor.[34] Among points covered were the following:

> "Because of the novel character of this installation, the Committee in its letter to the General Manager, dated July 29, 1955, felt obliged to point out that while the site had desirable features, and was considered suitable for a contained power reactor, certain safeguards should be provided and specific technology would have to be developed in order to give assurance of safe operation. These reservations were further amplified by the Committee in its letter of June 6, 1956, following a review conducted at its June 3, 1956 meeting.
>
> "In the years following the 1956 review, the applicant, the AEC and fast reactor groups abroad have conducted vigorous programs of developmental research which have greatly increased the knowledge of the behavior of fast reactors having high power densities. In particular fast reactors such as the Dounreay Fast Reactor in the U.K., BR-5 in the USSR, and two additional core loadings of the EBR-I have been operated for some time. This information has resulted in improved understanding and confidence in analytical techniques.
>
> "Specific information has been developed which allows a more accurate and reassuring evaluation of such important features of the reactor design as temperature and power coefficients, core distortions, Doppler effects, and the containment system. The Committee has followed these programs closely.
>
> "The reservations expressed by the Committee in its two earlier letters have been removed by the considerable body of data that has been amassed in the intervening years."

On August 6, 1962 the Power Reactor Development Corporation was allowed to extend the latest completion data of the Fermi reactor from September 1, 1962 to May 1, 1963.[35] The extension will allow time to complete construction and testing arising from developmental problems discussed earlier in Nuclear Safety.[36]

EBOR

On October 11, 1962, the ACRS reported on their review of the preliminary safeguards report of the 10-Mw(t) experimental beryllium oxide reactor (EBOR). This high-temperature gas-cooled beryllium oxide-moderator reactor is to be constructed at the National Reactor Testing Station. The ACRS suggested several additions to the safeguards proposed in the EBOR report. They were:

> "1. A reliable secondary shutdown system should be provided. A single failure could incapacitate all four safety rods. The liquid emergency cooling system does not fullfill the shutdown function, because it may not supply reactivity suppression.
>
> "2. Firm assurance should be provided that the addition of the emergency cooling solution will not produce significant damage to the reactor vessel by thermal shock.
>
> "3. The filtration and iodine removal system should include the building ventilation air. There would then be no need after an accident for special operator action to place the cleanup system in operation, and the chance of release of untreated air would be reduced.
>
> "4. The design should minimize the required operator actions in the event of an emergency."

The ACRS felt that with these modifications and with assurances to be provided by a proposed irradiation testing program, the reactor could be constructed at the proposed location "with reasonable assurance of subsequent operation with no undue hazard to the health and safety of the general public and the NRTS personnel."[37]

Hallam Nuclear Power Plant (Docket 115-3)

North American Aviation, Inc., was issued, on August 9, 1962, an amendment to its provisional operating authorization that will allow the 240-Mw(t) sodium-cooled Hallam reactor to be operated up to 15% of

full power.[38] On October 11, C. Starr, President of the Atomics International Division of North American Aviation, proposed that the AEC grant permission for full-power operation of the reactor without further hearings.[39] The proposal is under consideration. Starr also requested that the Consumers Public Power District be substituted for Atomics International as the reactor operator.

Humboldt Bay Reactor (Docket 50-133)

On August 28, 1962, the Pacific Gas and Electric Company was issued an 18-month provisional full-power operating license for its 163-Mw(t) boiling-water Humboldt Bay reactor.[20] The AEC Hearing Examiner had recommended issuance of the license in a decision of August 17.[40] After issuance of the license, the AEC received individual letters from J. B. Neilands and D. E. Personen of Berkeley, California, making motions "to file and exception and/or intervention" with respect to granting the reactor a license.[41] The Commission at a session on September 28 ordered that the motions be denied in all respects.[42] It was held that the petitioners had not appeared at the license hearings and were not parties to the proceedings. In addition it was felt that the requests were untimely and that there was no showing of interest that would justify intervention.

In other action of September 28 the Pacific Gas and Electric Company license was amended to grant an exemption from the requirements of Section 20.203(c)(2) of Part 20 of the Code of Federal Regulations concerning visible and audible alarms in high radiation areas.[42] Administrative procedures will guard against inadvertent entry into those high radiation areas where there are no control or alarm devices.

Indian Point Reactor (Docket 50-3)

The Consolidated Edison Company of New York on September 18, 1962 notified the AEC that they planned to establish some additional activities within the exclusion area of their 585-Mw(t) pressurized-water Indian Point reactor.[21] The activities involve functions associated with transmission, distribution, and commercial aspects of the Company in North Westchester County, New York. An additional 200 employees and 150 vehicles will be present on the site. In October, Consolidated Edison reported to the AEC on several changes in their test and operating procedures.[43]

Nuclear Merchant Ship N.S. SAVANNAH

In early November the AEC received from the States Marine Lines an operations analysis report covering the period May 1962 to November 1962 for the 69-Mw(t) pressurized-water reactor of the N.S. SAVANNAH.[44] After a review of the operating experience, the ACRS issued a report on November 14 that made the following points:[45]

> "Experience with the N.S. SAVANNAH has shown that it is very important to have available an auxiliary electrical power source in port. It is even more important to have a reliable auxiliary power source at sea or during port entry in event of a scram or other unusual circumstances. The Committee urges early action to provide a reliable auxiliary source of power, with propulsion reversal capability, adequate for safe handling of the ship in port and at sea.
>
> "The Committee would like to point out that the use of inert gas within the containment, a procedure necessary to diminish the fire hazard, has reduced drastically the possibility of carrying out maintenance within the containment and the possibility of making observations of the primary system performance at pressure and temperature. Experience at other reactors has indicated that this is an important facet of reactor safety. The Committee urges that all possible speed be employed to change the present control rod system to one

which will avoid the problems observed in the present system and which will permit adequate observation and maintenance within the containment.

"The Committee notes that the containment leakage and filter efficiencies are being checked frequently and recommends continued vigilance since performance has not always met specifications. The Committee believes that the proposed scheme for modification of the reactor compartment ventilation, cooling, and filtration system will be a desirable improvement to the effectiveness and safety of this system and urges its early completion.

"The ACRS was informed of the proposed plans to visit the ports of San Francisco, Long Beach, Los Angeles, Honolulu, Portland, San Diego, and Balboa before proceeding to Galveston. The N.S. SAVANNAH still has not had a long operational period and therefore decisions as to the ports visited and the docks selected should be made conservatively. In particular, where more than one suitable docking site is available in the same general area, the Committee believes that the more conservative should be chosen regardless of other considerations.

"In summary, the Committee believes that by taking all practicable precautions the N.S. SAVANNAH, prior to the Galveston overhaul [in early 1963], can enter ports found suitable under a conservative application of the Interim Port Analysis Factors without undue hazard to the health and safety of the public. However, the Committee recommends that only after the long term problems identified above have been rectified should consideration be given to any modification of the existing Interim Port Analysis except to clarify them or make them more conservative."

Pathfinder Reactor (Docket 50-130)

On September 6, 1962, a suggested regulatory schedule for the 203-Mw(t) boiling-water Pathfinder reactor was made public.[46] If the schedule is met, the public hearing for a reactor operating license will be held in March 1963.

Peach Bottom Reactor (Docket 50-171)

On August 29, 1962, a petition was filed with the U.S. Supreme Court for a Writ of Certiorari concerning the lower court public hearing

decision that granted a construction permit for the 117-Mw(t) high-temperature gas-cooled Peach Bottom reactor.[25,47] Specifically the Court is being asked to review the decision which denied L. W. Goldberg (the petitioner) an intervention request concerning the permit.

Piqua Nuclear Power Facility (Docket 115-2)

On August 23, 1962 a provisional operating authorization was issued for the 45.5-Mw(t) organic-cooled and -moderated Piqua reactor.[24] A hearing concerning the authorization had been held in June.

In early September an estimated $4,000 fire damaged the facility during preliminary testing of the reactor. No one was injured and no radioactivity was involved. The fire started when organic material leaked from a thermocouple weld on a main coolant line.[48]

Plum Brook (NASA) Reactor (Docket 50-30)

In late August 1962 NASA was granted an 18-month provisional operating license for its 60-Mw(t) light-water-cooled and -moderated Plum Brook research and testing reactor.[24] Operation up to full steady-state power was authorized.

Saxton Reactor (Docket 50-146)

The AEC on August 15, 1962 authorized operation of the pressurized-water Saxton reactor at steady-state power levels not in excess of 20 Mw(t).[49] On October 9 the Saxton license was amended so that Phase I of the reactor's research and development program could be undertaken.[22] The program has two major objectives: (1) to investigate the use of

boric acid in solutions as a neutron poison for chemical shim and (2) to investigate operation at specific powers up to 16 kw/ft of rod length.

Vallecitos Boiling-Water Reactor (Docket 50-18)

The General Electric Company on August 10, 1962 agreed to restrictions on the operation of the 50-Mw(t) Vallecitos boiling-water reactor. The restrictions, which were felt to be required because physical removal of a single control rod could cause the reactor to become supercritical when cold and xenon-free, were as follows:

> "A. The center section of the missile shield shall be fastened in place by welding of the locking pin and such welds shall be removed only upon written authorization of the Manager, Reactor Operations, obtained after the reactor has been unloaded to the extent necessary so that any control rod may be fully withdrawn or physically removed from the cold xenon-free core and the reactor maintained sub-critical.
>
> "B. Whenever any work is to be performed on a control rod drive involving major disassembly or removal of components, the reactor core shall be loaded such that any control rod may be fully withdrawn or physically removed from the core without the reactor reaching criticality and at least one of the 12" refueling port nozzle covers shall be removed to vent the reactor vessel to the containment atmosphere.
>
> "C. An audible alarm shall be provided near the control rod drive. This alarm shall sound at any time that one or more rod blocks are removed and the reactor pressure is below that at which any control rod may be withdrawn to its maximum full-out position or physically removed from the core without the reactor becoming critical. . . .
>
> "D. The key for the key-lock by-pass switch associated with the control rod stop system shall be kept in the custody of the Manager, Power Reactor Test Operation when not in use. Specific written procedures shall be prepared for each occasion on which the key is used.
>
> "E. The reactor will be operated with control rod configurations such that at any time during operation that any one of the control blades fails to move from its operating position of withdrawal when the scram signal is given, the reactor will remain sub-critical by 0.005 K, by the negative

reactivity due to temperature and/or xenon. If more than one of the withdrawn control rods fails to move or to travel to the full-in position when a scram signal is given, the reactor shall be maintained in a sub-critical condition by a minimum of 0.005 K by the use of the liquid poison system described in the application at such time as it is required to maintain this minimum shutdown margin. Poison must be injected after 24 hours even if temperature can be maintained and if all control rods have not been fully inserted or if some fuel has not been removed from the core."

On August 30, the ACRS reported on the VBWR shutdown margin as follows:[50]

"With the mechanical and electrical safeguards, and the administrative procedures as described in the referenced documents[49,51,52] made effective, and because the diminished shutdown margin will exist for a definitely limited time during which the consequences of excessive control rod withdrawal will be continuously recognized by operating personnel, the ACRS believes that the VBWR may be operated as proposed without undue risk to the health and safety of the public."

Vallecitos Experimental Superheat Reactor (Docket 50-183)

The General Electric Company on October 6, 1962 advised the AEC that the expected completion date of the 12.5-Mw(t) Vallecitos Experimental superheat reactor should be changed from February 28 to June 30, 1963.[43] The General Electric Company wishes to convert the construction permit to a 10-year operating license at an early date.

Yankee Reactor (Docket 50-29)

On August 25, the ACRS expressed their feelings on proposed changes at the pressurized-water Yankee reactor.[50] Comments on the changes, which entail increasing the maximum steady-state power level from 485 to 540 Mw(t), were as follows:

"This power plant has operated successfully throughout the life of Core I. The experience obtained during this period has indicated the possibility of operation at a higher power level with Core II. The major operational and plant

changes proposed include: (a) an additional safety injection pump, (b) automatic low pressure activation of the third charging pump, (c) changed scram requirements, (d) elimination of automatic rod withdrawal, and (e) a limitation of amount of rod withdrawal at power during any eight-hour period.

"Power coefficients and moderator temperature coefficients have been determined at 2000 equivalent full power hour intervals during Core I life to determine the possible effect of plutonium build-up. These tests were carried on to substantiate previous theoretical calculations which had indicated that the effect of plutonium build-up would not be significant. Data developed during Core I life have shown that there has been no significant effect. The licensee proposes now to make these determinations during initial start-up after fuel changes, and during scheduled generator load changes and plant shut-down only.

"The Advisory Committee on Reactor Safeguards believes that the change in frequency of the 2000-hour tests may be allowed and that this reactor may be operated at a power level of 540 Mw thermal without undue hazard to the health and safety of the public."

At a public hearing on September 20, the AEC Licensing and Regulation Division recommended that the proposed increase in power level be granted despite a recent setback at the reactor.[53] It was reported at the hearing that the ability to use 11 of the 22 flux wire thimbles in the reactor core had been lost and that the faulty thimbles had been sealed to prevent loss of coolant. It was felt that the reactor can be operated in complete safety with the remaining thimbles. Yankee also reported on recent sticking of one control rod. The difficulty had been corrected, however, and all 24 rods had gone through successful drop tests.

On October 11 Yankee was granted the license amendment authorizing the requested increase in power level and revising the technical specifications.[22] (J. R. Buchanan)

SAFEGUARDS REPORTS AND SELECTED READING

The recently issued safeguards reports and selected literature pertaining to hazards of reactors are listed below for reference. Because of the similarity of many reactors (in particular, research reactors), this list is not intended to be all inclusive.

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2. Babcock and Wilcox Co., Pool Test Reactor Hazards Evaluation, USAEC Report BAW-74, May 1958.
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4. J. Tadmor (ed.), Israel Research Reactor-I Hazards Evaluation Report, Israel Atomic Energy Commission Report IA-689, December 1961.
5. F. L. Bentzen and J. G. Crocker, SPERT IV Hazards Summary Report, USAEC Report IDO-16689, Phillips Petroleum Co., July 7, 1961.
6. R. E. Heffner et al., SPERT IV Facility, USAEC Report IDO-16745, Phillips Petroleum Co., Feb. 22, 1962.
7. A. Nelson Tardiff, Some Aspects of the WTR and SL-1 Accidents, USAEC Report IDO-19308, Apr. 9, 1962.
8. Los Alamos Scientific Lab., Fast Reactor Core Test Facility Safety Analysis Report, USAEC Report LA-2735, August 30, 1962.
9. J. E. Mahlmeister et al., Engineering and Constructing the Hallam Nuclear Power Facility Reactor Structure, USAEC Report NAA-SR-7366, Atomics International, 1960.

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12. Selwyn S. Berg, Final Safeguards Report to the U.S. Atomic Energy Commission for the Cornell University Zero Power Reactor, USAEC Report NP-12045, Cornell University, May 1962.

13. Hazards Summary for the L-77 Laboratory Reactor for the University of Nevada, Reno, USAEC Report NP-12048, Sept. 14, 1962.

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4. Duquesne Light Co., Reactor Plant Container Integrity Test, Core 1, Seed 3, Test Evaluation USAEC Report DLCS-3570105, May 7, 1962.

5. Duquesne Light Company, Monthly Operating Report, April 1962, USAEC Report DLCS-5000462.

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7. Duquesne Light Co., Monthly Operating Report, June 1962, USAEC Report DLCS-5000662.

8. Duquesne Light Co., Monthly Operating Report, July 1962, USAEC Report DLCS-5000762.

9. Phillips Petroleum Co., Organic Coolant Reactor Program Quarterly Report, January 1 – March 31, 1962, USAEC Report IDO-16787, June 29, 1962.

10. O.E.C.D. Halden Reactor Project, Third Annual Report, Norwegian Report NP-11912, 1961.

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12. Duquesne Light Co., Monthly Operating Report, February 1962, USAEC Report TID-15224.

13. Westinghouse Electric Corp., Pressurized Water Reactor (PWR) Project Technical Progress Report, April 27, 1962 to June 23, 1962, USAEC Report WAPD-MRP-98.

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