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THE OAK RIDGE RESEARCH REACTOR – A FUNCTIONAL DESCRIPTION

T. P. Hamrick
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OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION

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OPERATIONS DIVISION

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SEPTEMBER 1968

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PREFACE

This description of the Oak Ridge Research Reactor (ORR) is intended to serve the twofold purpose of presenting a reasonably comprehensive picture of the system as a basis for other more specialized studies and of providing the necessary background material for the "Safety Analysis." The description presented here reflects the design as it was in January 1967.

This report is basically descriptive in nature and is for the purpose of acquainting the reader with just what the ORR is and how it operates. Except where necessary for clarity, design calculations are not included. The interested reader will find in Appendix E a bibliography of ORR reports that go into more detail on the many aspects of design and operation.

The information contained herein has been compiled from reports of studies performed by the various members of the ORR Project during the days of construction and by Operations Division personnel during the nine years of operation since March 1958. All the persons who contributed are too numerous to list, but special mention is made of F. T. Binford, C. D. Cagle, S. S. Hurt III, and W. H. Tabor for their guidance in the preparation of this report.

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T. P. Hamrick J. H. Swanks

1. INTRODUCTION

1.1 Historical Background and Motivation

The history of the Oak Ridge Research Reactor (ORR) began in the early part of 1950, when it was determined that a higher-flux reactor facility was required at ORNL in order to successfully carry out the various research programs then being undertaken. The reactor originally selected was a simplified version of the Materials Testing Reactor (MTR) to operate at a power level of 5 Mw, which would require a minimum of design and development. During the initial phase, the proposed design changed considerably as required by the shift in emphasis on the experimental programs. With the operating experience of the MTR and other reactors as a guide and the additional data available in the field of reactor technology, a number of improvements were made in the basic design.

The need for improved facilities for experiments requiring neutron beams, for expanded isotope production, and for an irradiation facility to permit engineering tests of the materials and components of liquid fuel reactors became increasingly important. Construction and operation of the ORR helped meet the requirements for neutron irradiation space of both the basic and applied research programs of the Laboratory and met the expanding requirements for radioisotope production at ORNL.

A preliminary proposal which outlined the basic features of the desired reactor was submitted to the AEC in March 1954. This proposal was approved, and a directive was issued permitting the use of funds to retain an architect-engineer to prepare a reactor design on the basis of 5-Mw operation. A contract, calling for the definitive design of the building, reactor shielding structure, and cooling systems, was executed in July 1954 with the McPherson Company of Greenville, South Carolina. In August 1954, a modification of the AEC directive was issued authorizing the design and procurement of the reactor and controls by ORNL.

In December 1954, a review was made of the status of the ORR Project coupled with a review of the requirements for irradiation space. One result of this review was the decision that sufficient justification existed to warrant an increase from the 5-Mw power level to 20 Mw. This increase in power level was discussed with members of the Advisory Committee on Reactor Safeguards; and in March 1955, the following design modifications were agreed to constitute adequate safeguard measures considering the design of the reactor, the location, and the proposed use:

1. Gas containment was provided to an extent that an accident releasing the volatile fission product gases from the reactor core would not constitute a widespread hazard.
2. Available excess reactivity was to be minimized insofar as was practical.
3. The effects of experiments installed in the engineering test facilities would be limited so that upon failure or malfunction of the installation, no more than 1.4% reactivity could be added to the reactor.

A modification of the original directive was then issued providing additional funds to allow design and installation of equipment necessary for operation at 20 Mw. Construction of the building, reactor shielding structure, and cooling system was performed by Blount Brothers Construction Company of Montgomery, Alabama. This contractor was selected on a competitive lump-sum-bid basis.

Construction was completed in the spring of 1958, and the reactor achieved criticality on March 21, 1958. At the time the ORR was designed, air-cooled heat exchangers were provided to give 20 Mw of cooling capacity. Tests made during the summer of 1958 indicated that the capacity of the coolers was approximately 25% below their specified capacity. This meant that during the summer months the reactor was operated at 15 Mw, with 20-Mw operation possible only during the winter months. In many cases, experiments had been designed for a steady 20-Mw power level and, when power was reduced, could not be operated at the desired temperatures or neutron fluxes.

To provide experiments with a higher neutron flux and to realize more fully the potential capacity of the ORR as a research tool, it was proposed in September 1959 to increase the power level of the reactor by adding evaporative water cooling to provide as much as 30 Mw of heat-removal capacity.

The proposed power increase involved mainly the evaluation and/or establishment of the following: (1) increase in reactor primary coolant flow; (2) operation of the fuel plates at higher temperatures; (3) reduction of the uncertainties in the temperatures of hot spots which limit the power; (4) assurance that increased stress or heating would not damage the reactor tank or other structures; and (5) provision for emergency pumping capacity to cool the fuel in the event of an electrical power failure.

Permission to raise the power to 30 Mw was obtained from AEC, and modifications were completed in the summer of 1960. Since August 5, 1960, the ORR has been operated at a power level of 30 Mw.

1.2 Brief Description of the ORR

One of the unique features of the ORR is the location of the reactor tank in a pool of water. The water provides the necessary shielding for working above the reactor core and also makes access to the core as convenient as it is in low-power pool-type reactors. The majority of experimenters desire to place their experiment rigs in the very high-flux regions in the core and the reflector (Fig. 1.1); hence, an easy access to the core was stressed during the design. The control-rod drive mechanisms are located below the core, and the upper grid plate which is usually present in this type of reactor was replaced by two small, independent grids (Fig. 1.2). This last feature makes it possible to leave experiments in the core region while the reactor is being refueled.

The reactor core is a heterogeneous type which uses enriched uranium fuel in the form of aluminum-clad aluminum-uranium alloy fuel plates. A fuel element consists of an assembly of 19 fuel plates. Demineralized water serves as the reactor coolant and moderator. The reflector is composed of an arrangement of beryllium elements which are physically interchangeable with each other and with the fuel elements. The thickness of the reflector therefore varies according to the particular fuel and experiment arrangement in the core. About 4 ft of ordinary water surrounds the reflector. Figure 1.3 shows a cutaway view of the reactor core.

A rectangular aluminum box surrounds the core and beryllium reflector, and a grid plate is located below it to provide for the spacing and support of the fuel and beryllium elements. Fuel elements, control rods, experiments, and reflector elements are positioned in a 9 by 7 array by the grid plate (Fig. 1.1).

The reactor is controlled by vertically positioning control rods which are located in the fuel and reflector regions of the reactor core. Facilities for up to 12 control rods are provided, but only 6 of these positions are used. The poison sections of the control rods are approximately as long as the reactor core height and consist of an aluminum-jacketed cadmium sheet formed into a rectangular box. The lower half of the rod consists of a fuel section of the same composition and general type as the fuel elements, except that only 14 fuel plates are included in the assembly. The control rods may be used as combination shim-safety-regulating rods, and the follower sections may be chosen according to the amount of fuel or reflector material desired.

The heat-transfer calculations for the ORR were originally performed considering initial operation of the reactor at 20 Mw and possible future operation at the present power level of 30 Mw. Coolant flow requirements were established as 12,000 gpm for 20-Mw operation and 18,000 gpm for 30-Mw operation. A conservative design criterion was used which assumed the simultaneous occurrence

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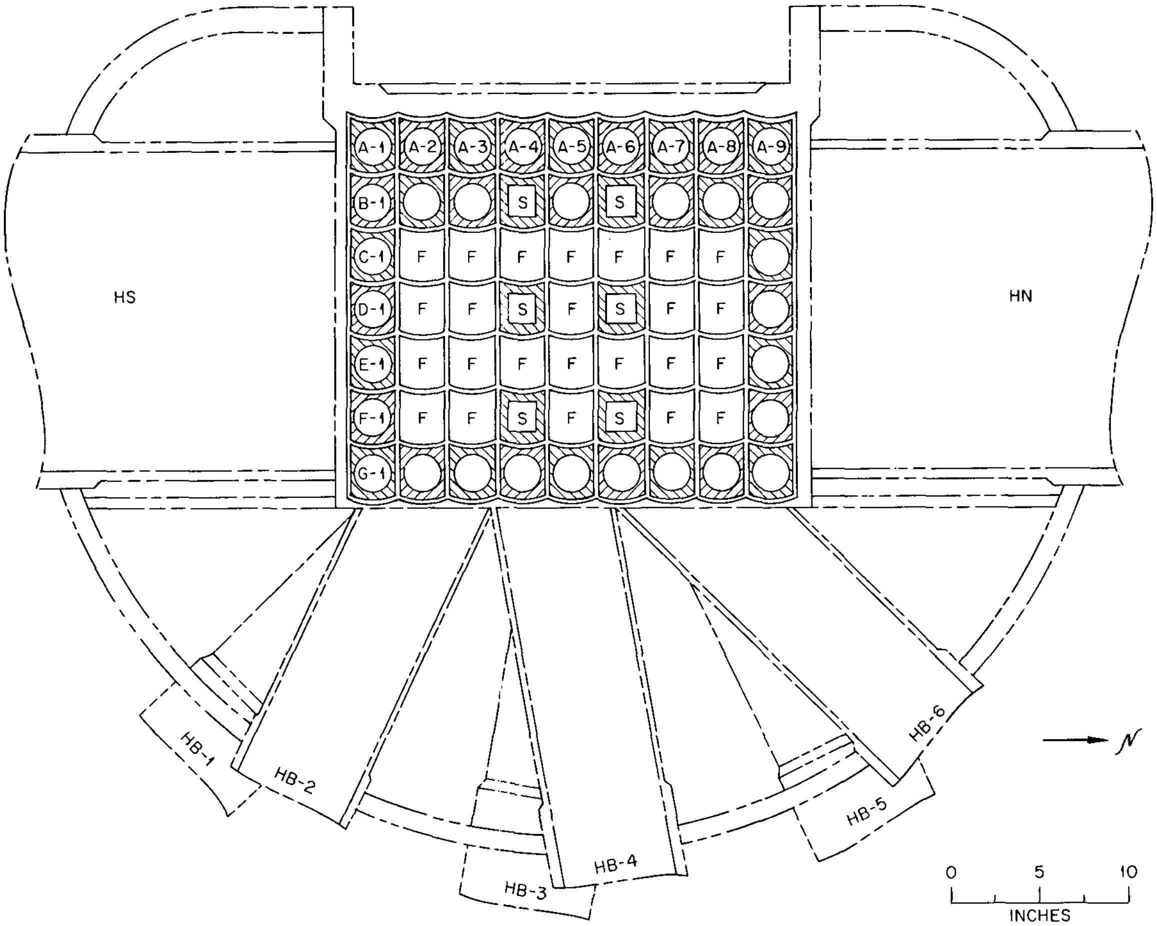

LATTICE PATTERN: F-FUEL S-SHIM ROD OTHERS-REFLECTORS
EXPERIMENT FACILITES: HN AND HS-LARGE FACILITIES HB-BEAM HOLES

Fig. 1.1. Horizontal Section Through Reactor Center Line.

of the worst possible mechanical, nuclear, hydraulic, and thermal conditions in the same physical location. Data from operating experience of the MTR were used as an aid in establishing criteria and as a basis for comparison of calculations with experimentally observed quantities. Hydraulic tests were performed on fuel elements, control rods, and reflector elements as an integral part of the design effort.

The cooling requirements for power operation of the ORR are provided by two separate cooling systems. One of these, the reactor cooling system, is designed to remove the ~30 Mw of energy produced in the core.

In this system demineralized water as the primary coolant is pumped through the reactor tank at a flow rate of about 18,000 gpm. It passes through the shell side of four heat exchangers, where it transfers its heat to the secondary coolant, which is circulated through the tube side of the heat exchangers. The secondary coolant, treated process water, is circulated through a conventional induced-draft cooling tower, which dissipates the heat to the atmosphere.

Approximately 0.6 Mw of reactor heat is transferred to the reactor pool by conduction from heated surfaces and by absorption of radiation. To dispose of this and up to 0.1 Mw of heat released by used fuel elements stored in the pool, a second cooling system, the pool cooling system, circulates 700 gpm of pool water through the tube side of a heat exchanger. Here the secondary coolant, treated process water, absorbs the heat while circulating through the shell side of the heat

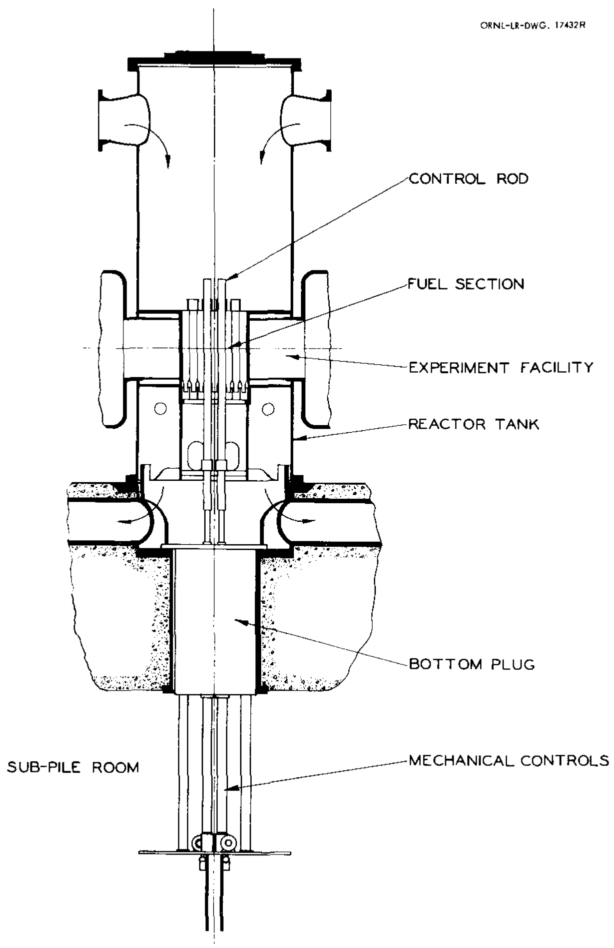

Fig. 1.2. Reactor Vertical Section.

exchanger. This secondary coolant is then circulated through a separate conventional induced-draft cooling tower, which dissipates this heat to the atmosphere.

Because of the heat generated by the fission product inventory in the core, it is necessary to cool the core for a short time following shutdown. Under normal circumstances, this is done by the normal cooling system, but in the event of a failure of power to the main pump motors, adequate coolant flow is maintained by battery-powered dc motors attached to the shafts of the main coolant pumps. Any one of these three small motors can provide a coolant flow sufficient to prevent core damage due to afterheat.

In a reactor cooling system, the radioactivity level, corrosion rate, and deposit formation rate must be closely controlled. At the ORR, these objectives are met by demineralizing the cooling water with multiple-bed ion exchange columns. There are two reactor demineralizer systems, one on stream and the other on standby, regenerated and ready for service. Both can be used simultaneously if conditions warrant it, for example, after a long shutdown, when the water may contain more impurities than during normal operation.

It is essential that the primary water have as few impurities as possible, since it passes through the high-neutron-flux core region, where impurities become highly activated, so some cooling water is continually bypassed through the demineralizers to remove trace impurities. Most of the components in the reactor cooling loop are made of aluminum, although some of the smaller items are stainless steel. Corrosion products, dissolved gases, and fission products which result from surface uranium contamination on the fuel plates constitute most of the impurities found in the primary cooling loop.

The radioactivity in the ORR pool water would also build up to an appreciable level if a method were not provided to remove radioactive ions. The neutron flux in the pool just outside the reactor tank wall is on the order of 10^{13} neutrons cm^{-2} sec^{-1} and produces radioactive ^{24}Na, ^{16}N, and other unstable nuclides; therefore, a bypass demineralizer is used to remove these nuclides from the pool water.

An additional demineralizer unit, consisting of separate anion and cation columns, is used in the reactor primary water system on the effluent from the degasifier. The primary function of this system is to decontaminate any water that expands from the reactor water system into the pool. An additional advantage is extra demineralizer capacity for the reactor water system.

The ORR bypass degasifier is designed to remove entrapped or dissolved gases from the water in the reactor primary cooling system. The air radioactivity in the reactor building was reduced significantly when the degasifier was installed.

In addition to the degasifier the ORR gas-removal system includes several ball-float traps, which are located above, and connected to, parts of the reactor water system where gases naturally collect. These ball-float traps, normally full of water, contain valves which open when gas is collected, thus allowing the gas to bleed into the interconnected off-gas system.

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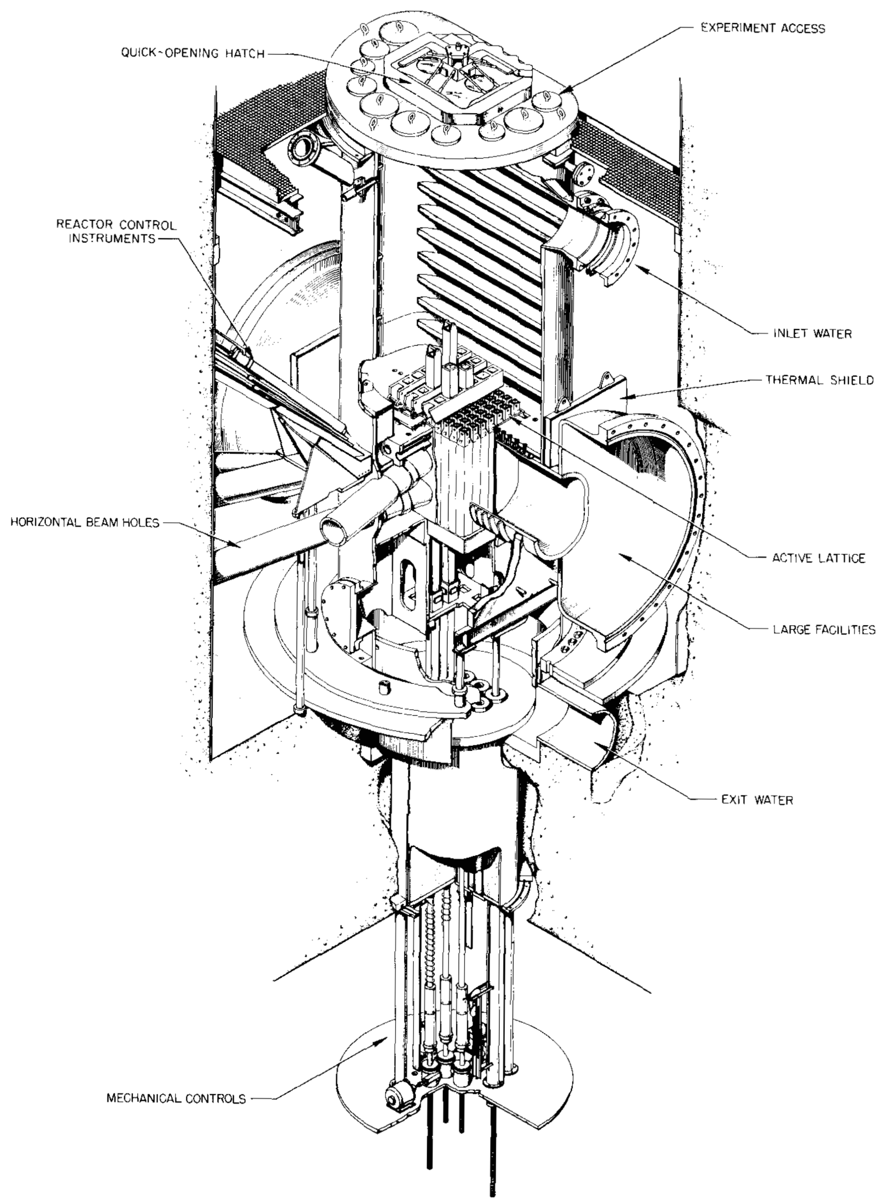

Fig. 1.3. View of the ORR Core and Tank.

A continuous supply of demineralized water is necessary to supply the shim-rod-drive seals and the bottom-plug seal in the subpile room, to service those reactor experiments for which an uninterrupted supply of demineralized water is required, to supply the reactor pools when makeup is needed, and to replace water lost from the primary system through cooling and sealing of various bearings and glands. Flow diagrams of the reactor and pool cooling systems are shown in Fig. 1.4 and 1.5.

The ORR safety and control systems have been designed to provide for safe and orderly operation of the reactor from a central control room. Essentially all routine operations, including startup and shutdown, can be monitored and/or controlled from this location.

The control system is designed to relieve the operator of routine manipulations by enabling the instruments which sense changes in the system parameters also to initiate the required corrective

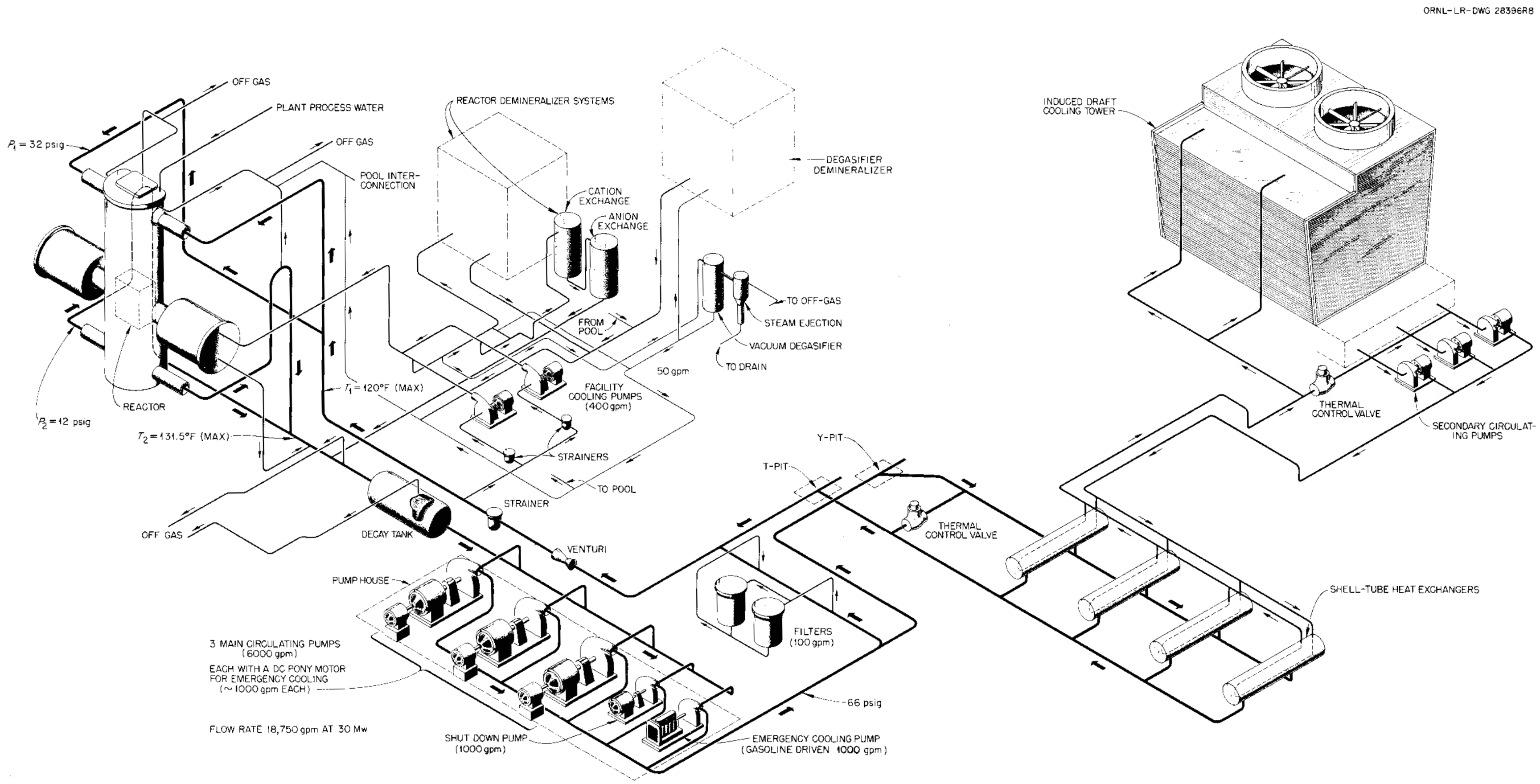

Fig. 1.4. Reactor Cooling System.

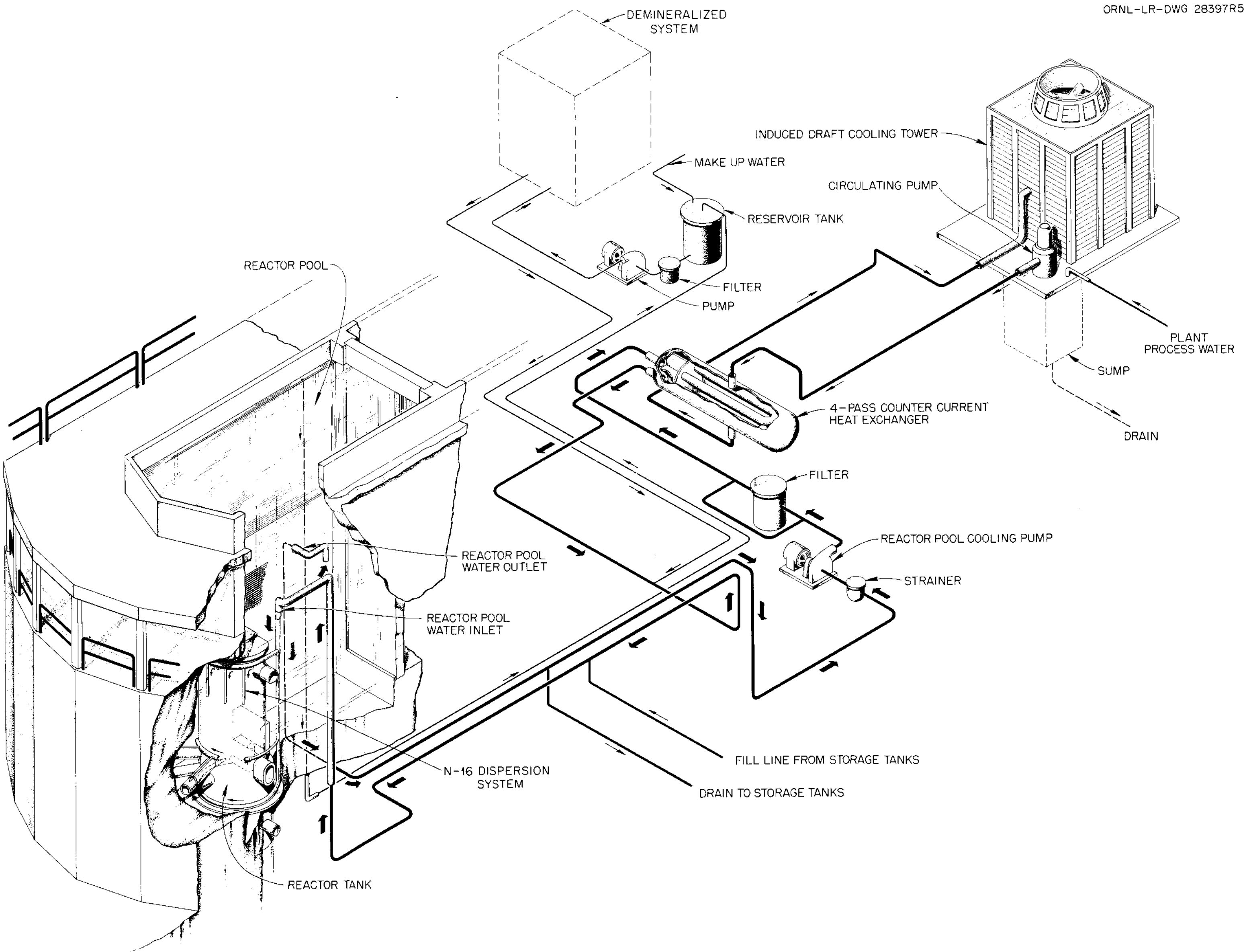

Fig. 1.5. Pool Cooling System.

action. This approach is consistent with the philosophy of using the operator to supervise the functions of the control system rather than to include him as an integral part of it. Nevertheless, certain actions are required of the operator. In particular, any increase in reactivity beyond that allotted to the power regulation system will require concurrence of both the operator and the control system. The safety system is designed to seize the initiative from both the operator and the control system and to initiate immediate corrective action should any of the significant operating parameters indicate the onset of an unsafe condition. Because of certain features incorporated in the control system, the safety system is very infrequently called upon; however, it is independent of the control system and capable of very fast response when needed.

The ORR control system and instrumentation were designed and safety limits determined after careful analysis of pertinent reactor parameters such as fuel loadings, shim-rod worth and motion, moderator, coolant, etc. This design resulted in the use of instrumentation similar to that installed in both the LITR and the MTR and, in addition, contains a provision for an automatic start mode.

Process instrumentation monitors reactor coolant flow, reactor coolant differential temperature, and reactor coolant outlet temperature. Signals from these monitoring instruments initiate a power reduction by setback, reverse, or slow scram as dictated by the relative magnitude of the error signals. Reactor coolant temperature control during a reactor power increase is performed automatically by step-function control of the reactor cooling system (Fig. 1.6).

The experiments in the reactor are provided with reactor power-reduction circuits, if required. These circuits are capable of initiating a setback, reverse, or slow scram if a particular experiment parameter should exceed preset control limits.

As previously pointed out, one of the main objectives of the ORR operation is to continuously maintain the reactor at full power level as long as this is consistent with safety requirements such as the available coolant flow. To accomplish this, heat power has been chosen as the basic control parameter, although it is used indirectly. The instruments which measure flow and temperature difference are quite accurate but are characterized by an inconveniently long response time – several seconds. Even though an exceptionally fast response time is not intrinsically required of the control system, it is desirable that it be able to initiate corrective action sufficiently rapidly so that fast safety action will be only rarely necessary. The required speed of response is obtained by using the accurate, but delayed, heat-power information to continually calibrate neutron-flux measuring devices by adjusting them to agree with the heat-power information. Block diagrams of the ORR control system are shown in Fig. 1.7 and 1.8.

The shielding was designed to satisfy the permissible radiation dose limitations to individuals established by the National Bureau of Standards for a steady operating power of 30 Mw. In unlimited access areas, the shield design is such as to limit the maximum dose rate to 0.75 millirem/hr. This intensity includes the combined effects of all radiations, and, based upon a 40-hr week and a 50-week year, it represents 30% of the annual permissible dose of 5 rems recommended by the NBS handbooks as an acceptable amount for workers handling radioactive materials. In limited access areas, sustained exposure is unlikely and can be administratively controlled, so higher radiation levels are permitted there. This fact makes possible some economy in shield design.

Adequate shielding is provided not only for the reactor itself but for portions of the primary coolant loop and for the primary and pool coolant cleanup systems as well. Shielding is also provided where necessary for the various components of the off-gas and ventilation systems. The demineralizers, filters, degasifier, and other equipment are also located in individually shielded cells or cubicles. Shielding provided for the cells is supplemented in some cases with direct lead shielding on the equipment.

The ORR employs the concept of dynamic containment to prevent the escape of radioactive gases to the atmosphere. The ORR building is not sealed airtight to retain radioactive gases that might be released. Instead, it is maintained as a partially leak-tight structure. The cell ventilation system maintains the containment by the controlled exhaust of air from the building at a rate sufficient to ensure that there is always an inflow of air at leakage points. The cell ventilation system is not a start-on-demand system, but rather it operates continuously, so that the building is always under a slight vacuum.

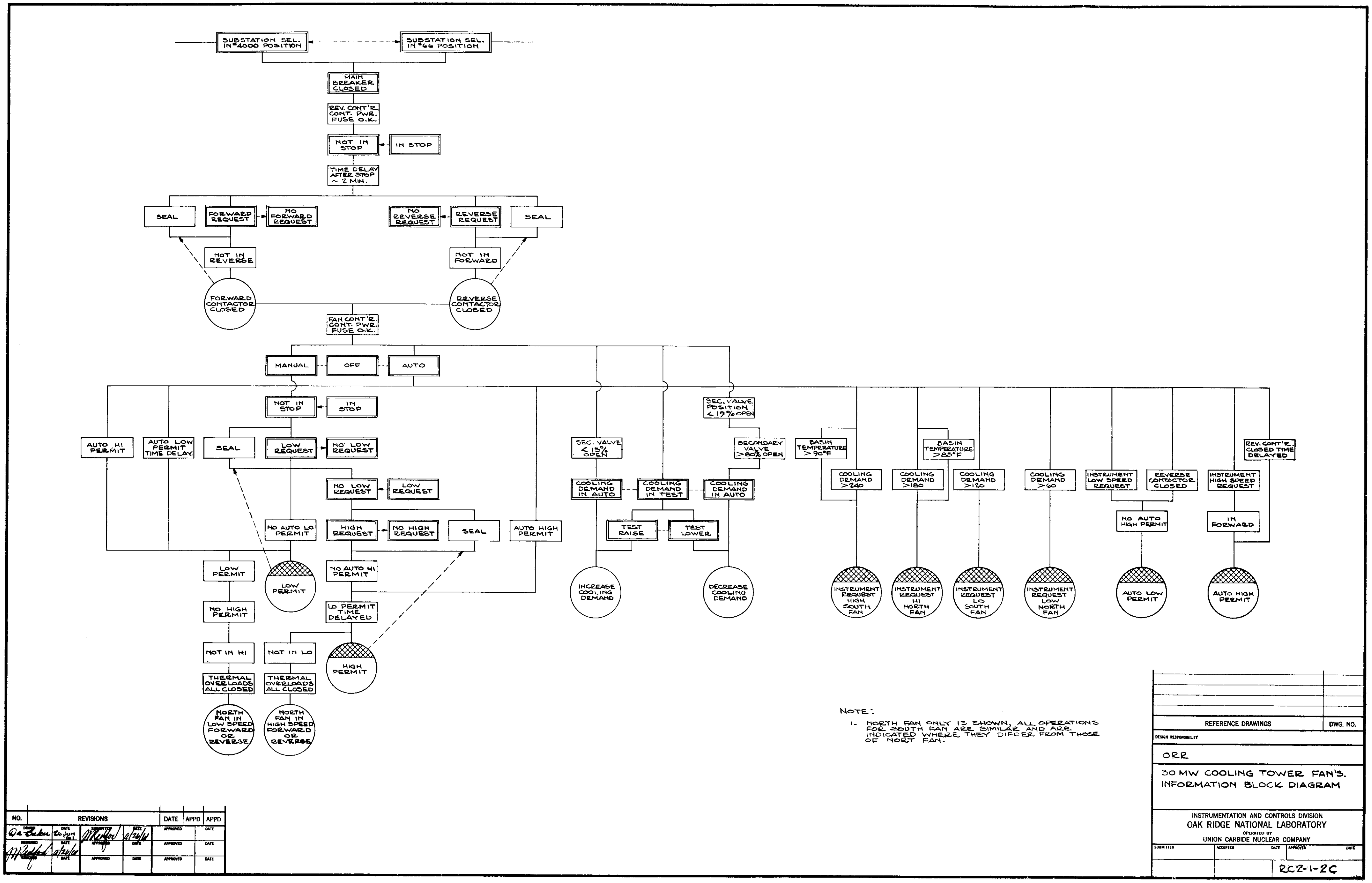

Fig. 1.6. Cooling Tower Control Block Diagram.

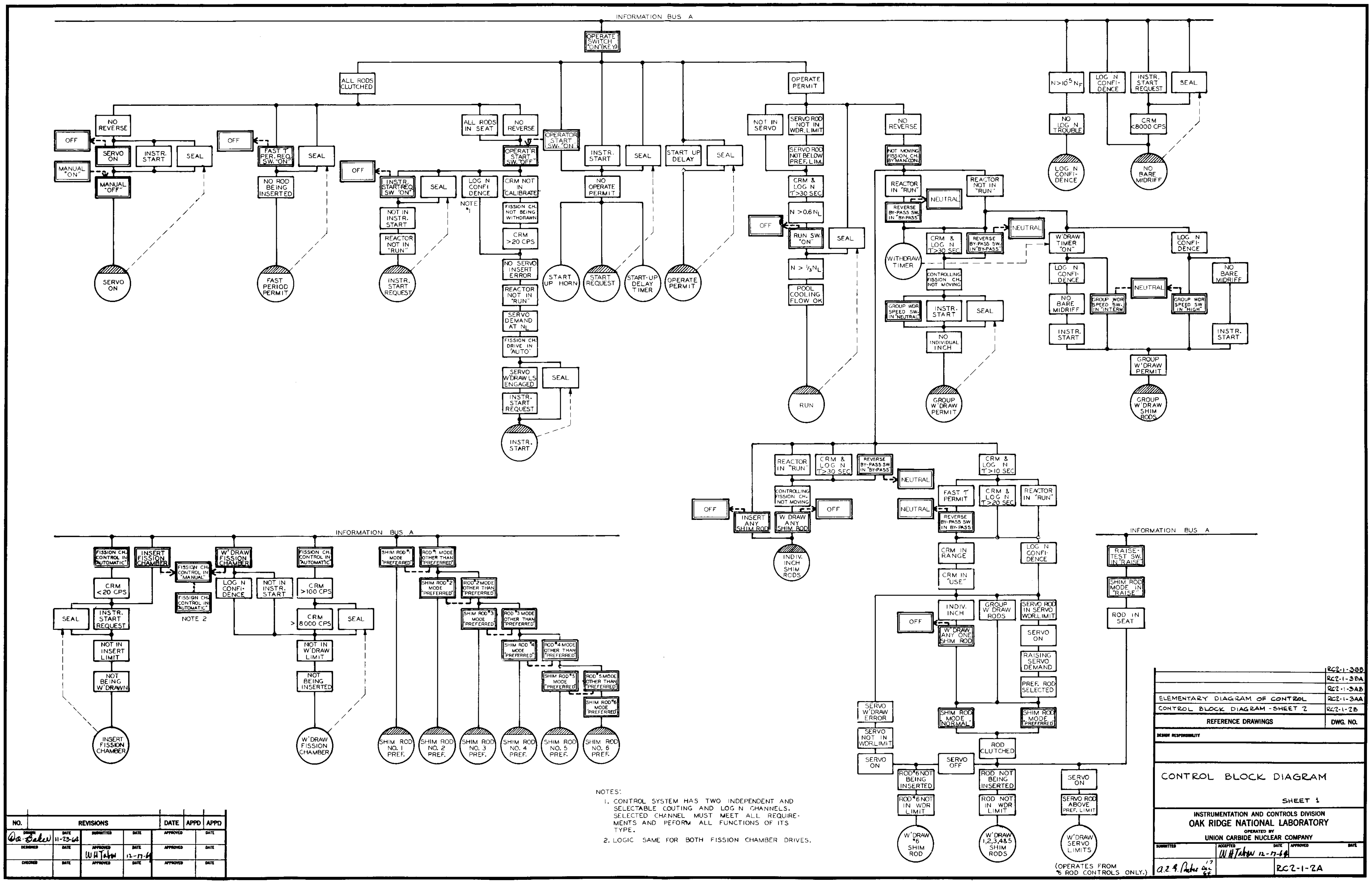

Fig. 1.7. Reactor Control Block Diagram.

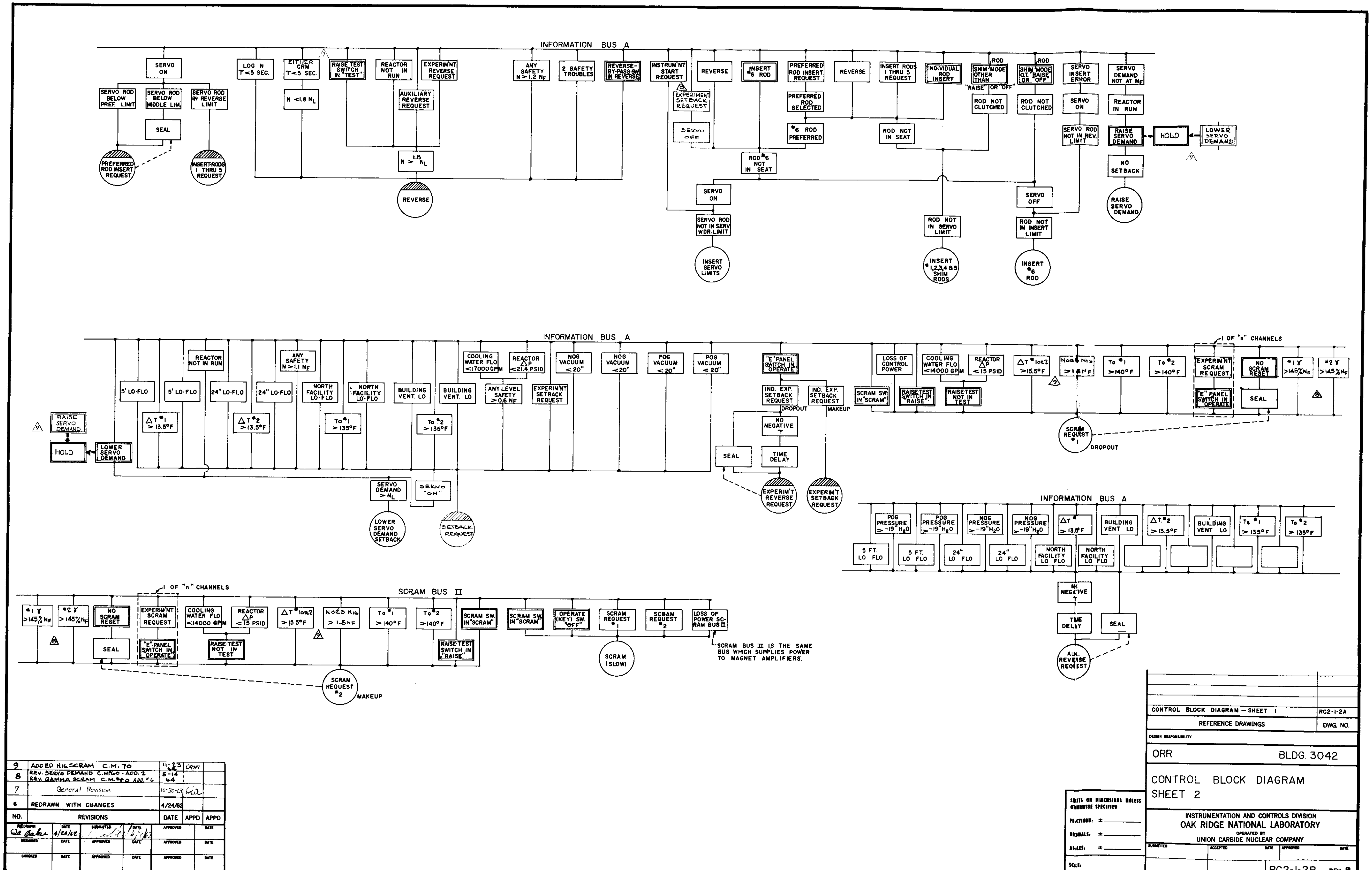

Fig. 1.8. Reactor Control Block Diagram (Continued).

The Oak Ridge Research Reactor is one of 17 facilities at Oak Ridge National Laboratory equipped with networks of radiation and air monitoring instruments connected to the Laboratory Emergency Control Center. The network in the ORR building is also monitored from the ORR control room. Radiation and Contamination Detection Systems are installed in the reactor building to continuously and automatically determine the radiation condition of the entire facility and to relay this information to the central control panelboard in the ORR control room. Should the radiation level or the air activity in a large portion of the building exceed preset values, an audible alarm in the building is actuated, warning lights outside the building flash, and an alarm signal is transmitted to the Laboratory Emergency Control Center.

Complete descriptions of the individual components of the system are found in appropriate sections of this report.

2. REACTOR SITE

2.1 Location

The ORR and its ancillary facilities are located in the Roane County portion of the AEC reservation at Oak Ridge, Tennessee, and are shown on maps of increasing scale in Figs. 2.1 to 2.6. The ORR is located in the ORNL, X-10 site. It is 100 ft directly east of Building 3001 (the ORNL Graphite Reactor, now shut down), 50 ft directly south of Building 3005 (the Low-Intensity Testing Reactor), and 100 ft southwest of Building 3010 (the Bulk Shielding Reactor).

The ORR is located within the well-established ORNL controlled-access area. Adequate personnel- and visitor-control policies have been established so that only necessary operating personnel and persons having legitimate business are permitted within the immediate area around the ORR. Approximately 70 people are present in the reactor building during normal day-shift hours with only 4 people normally required for off-shift operation.

2.2 Population Density

The total population of the four counties (Anderson, Knox, Loudon, and Roane) nearest the ORR site is 370,145. Of this number, 177,255 are located in cities with populations of more than 2500 persons. The rural population density in these four counties is about 135 persons per square mile. The average population density within a radius of 27.5 miles of the ORR site, as determined from the data in the 1960 census, is 147 persons per square mile. Table 2.1 lists the populations of the surrounding communities which have a population of over 500, together with their distance and direction from the ORR site. The rural population density in the four surrounding counties and in two other nearby counties is given in Table 2.2. A number of plants are located within the AEC-controlled area and nearby; the approximate number of employees located in each plant is given in Table 2.3. These data indicate the total employment at each facility and do not attempt to show the breakdown according to shifts. However, practically all of these employees work the normal 40-hr week.

An estimate has been made of the distribution of the resident population in each of the 16 adjacent $22\frac{1}{2}$-deg sectors of concentric circles originating at the ORR. Eight different incremental distances from the ORR site were considered: 0–0.5, 0.5–1, 1–2, 2–3, 3–4, 4–5, 5–10, and 10–20 miles radii. The estimated resident population distribution is given in Table 2.4. These data are representative of the population in this area at all times. Very little variation is experienced owing to either part-time occupancy or seasonal variation. Population density in the area has been reasonably stable for a number of years and is expected to remain so.

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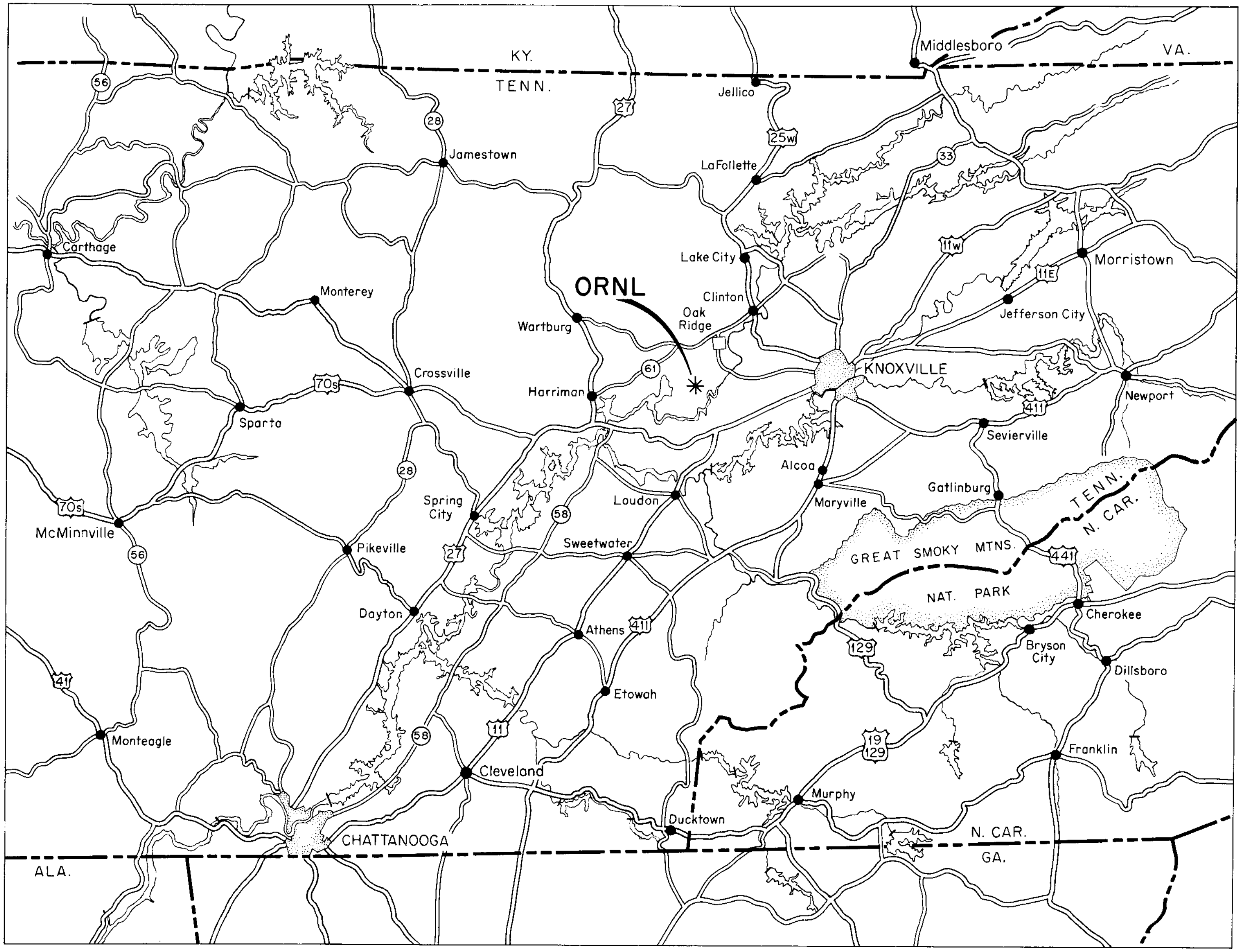

Fig. 2.1. Area Within 100 Miles of Site.

ORNL-LR-DWG 38362AR

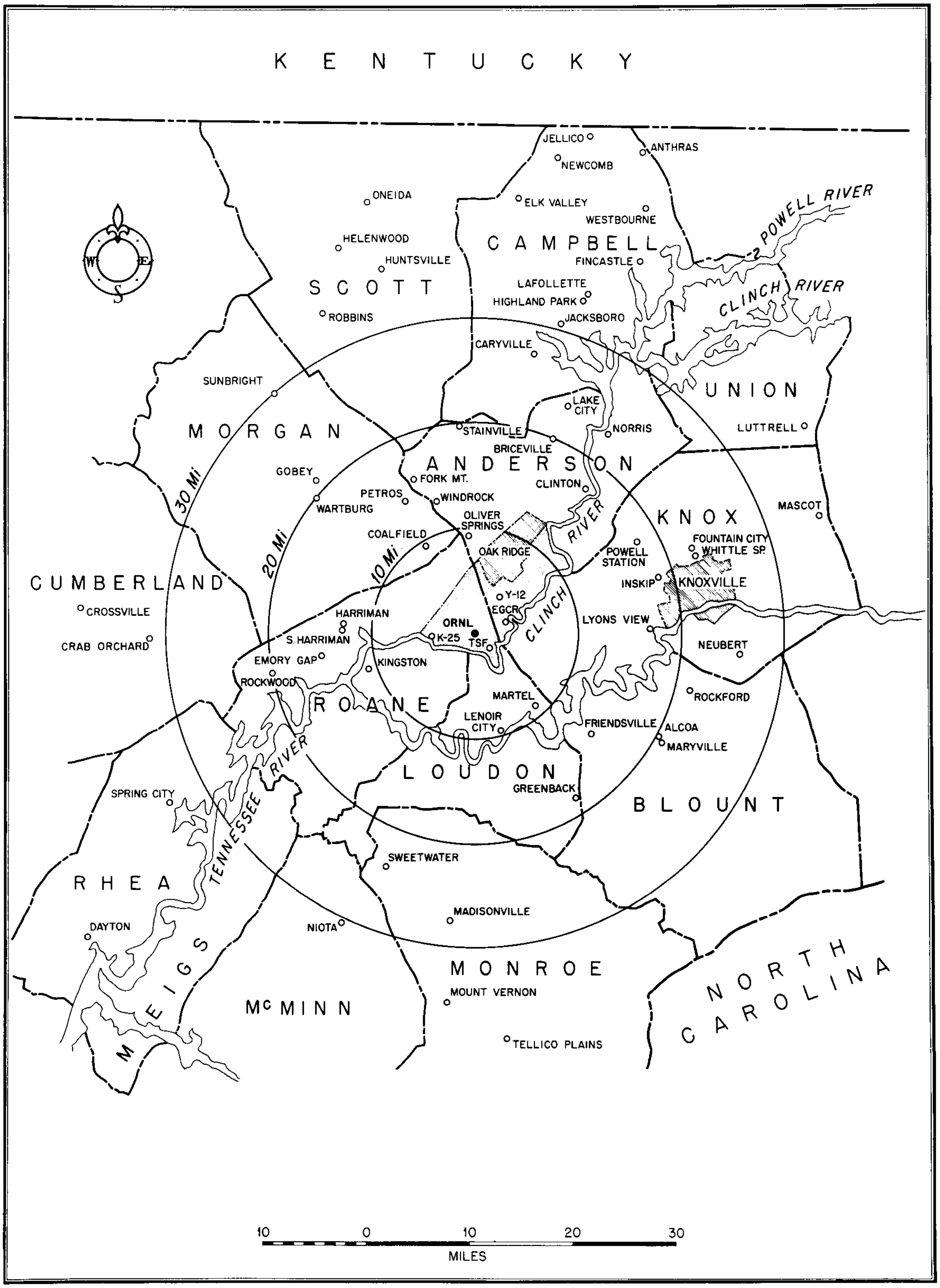

Fig. 2.2. Map of Cities and Counties Surrounding Oak Ridge Area.

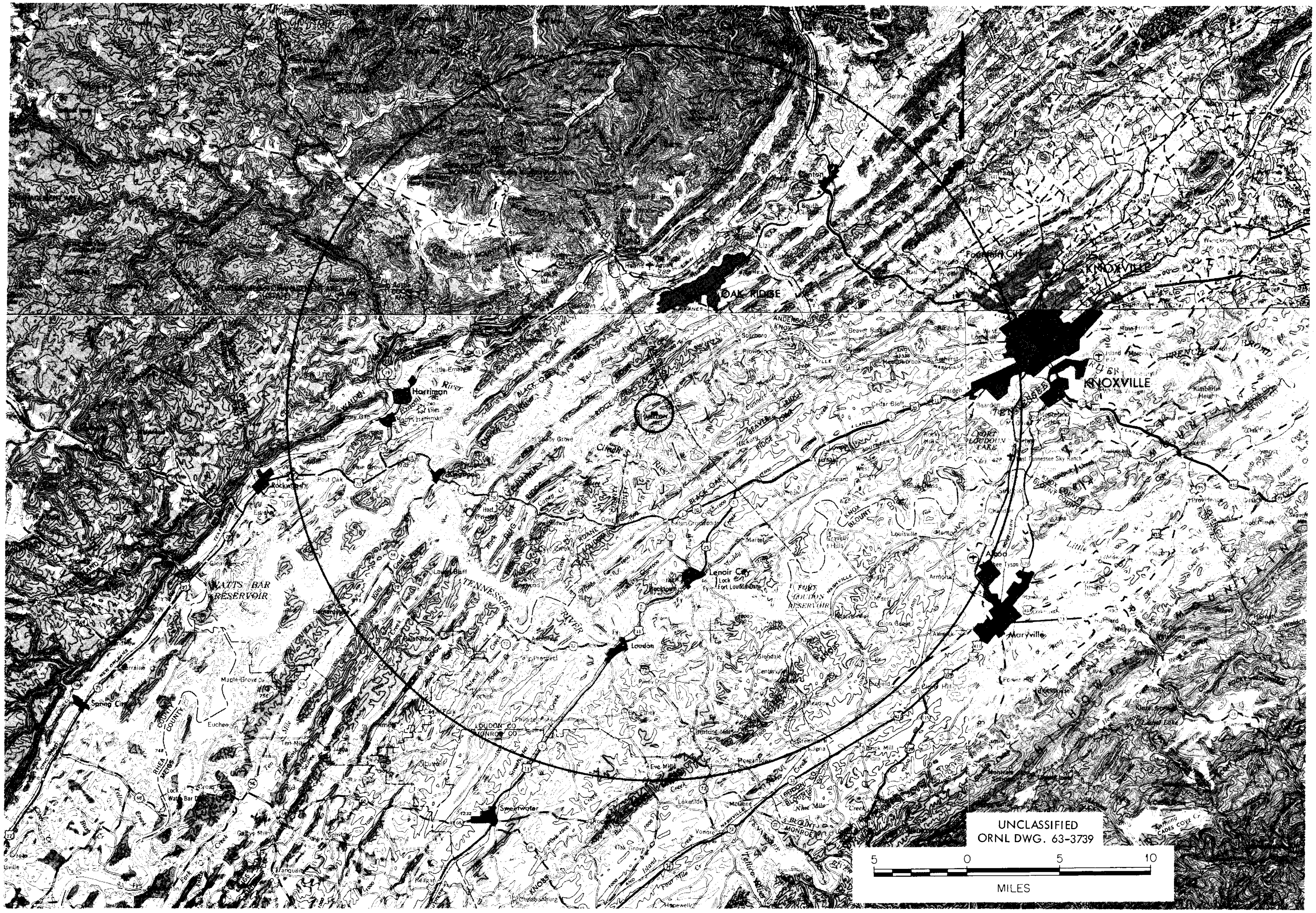

Fig. 2.3. Twenty-Mile Radius Circle of Site.

ORNL-DWG 63-3744

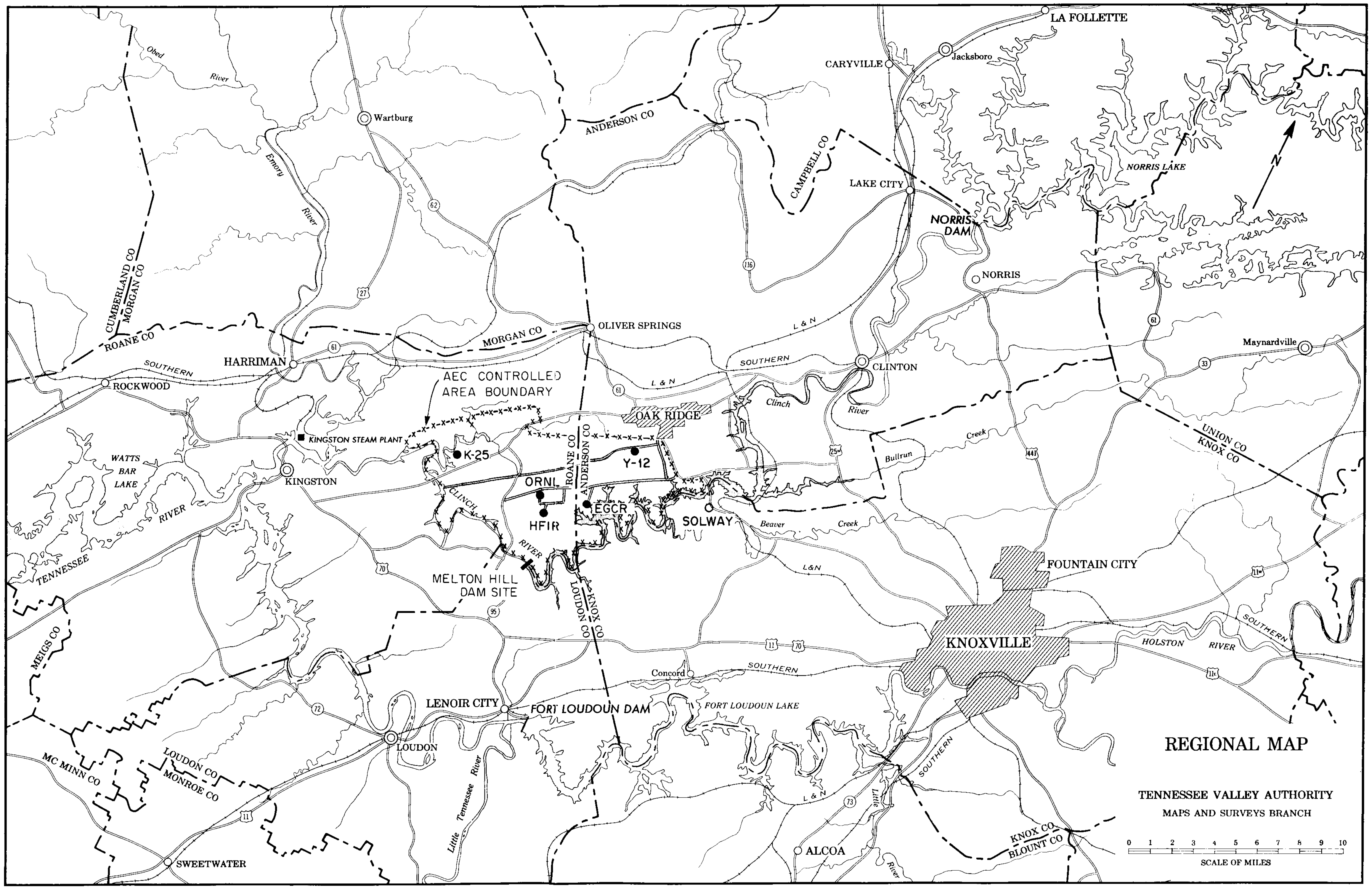

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Fig. 2.4. Regional Map.

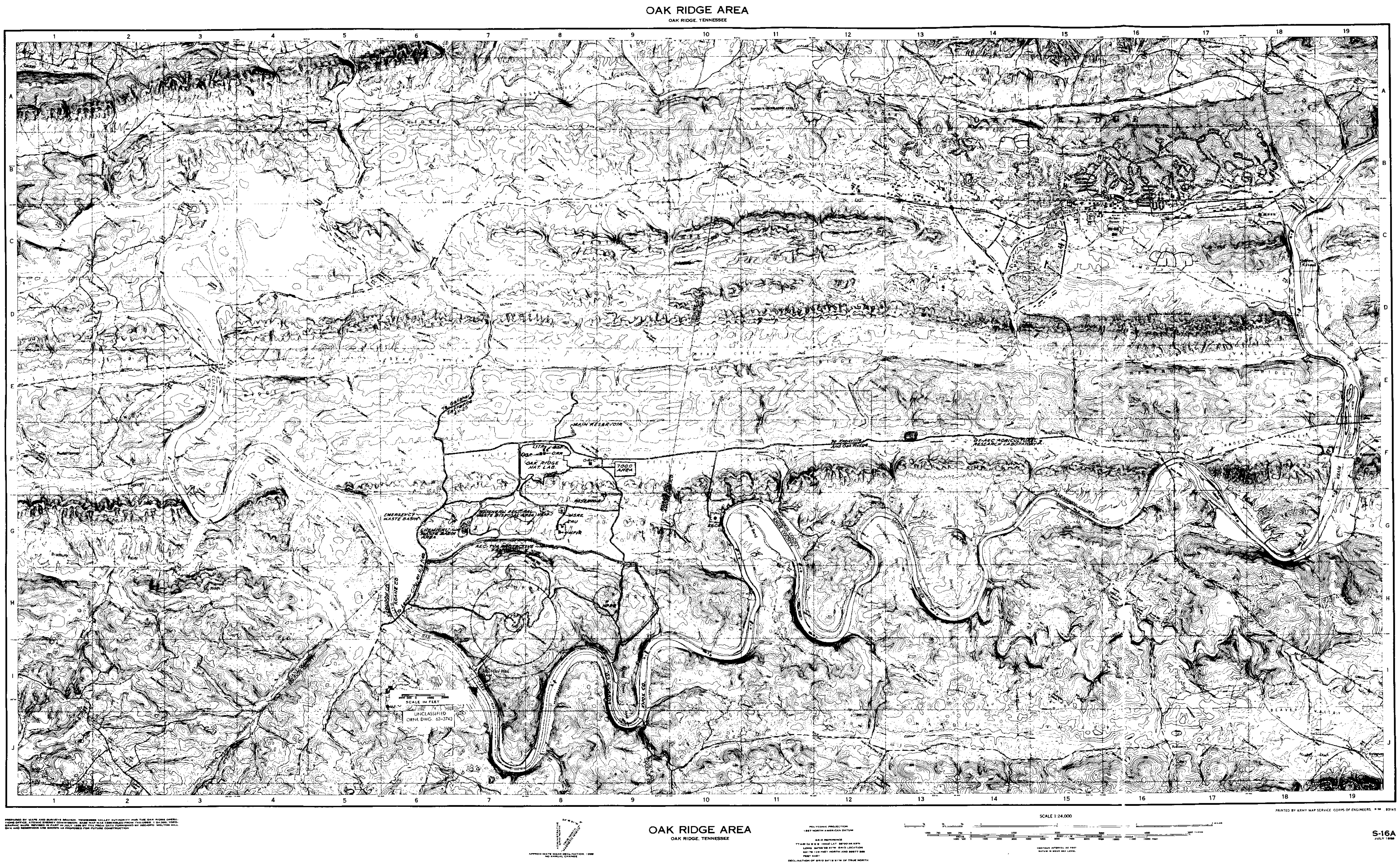

Fig. 2.5. Contour Map of ORNL Area.

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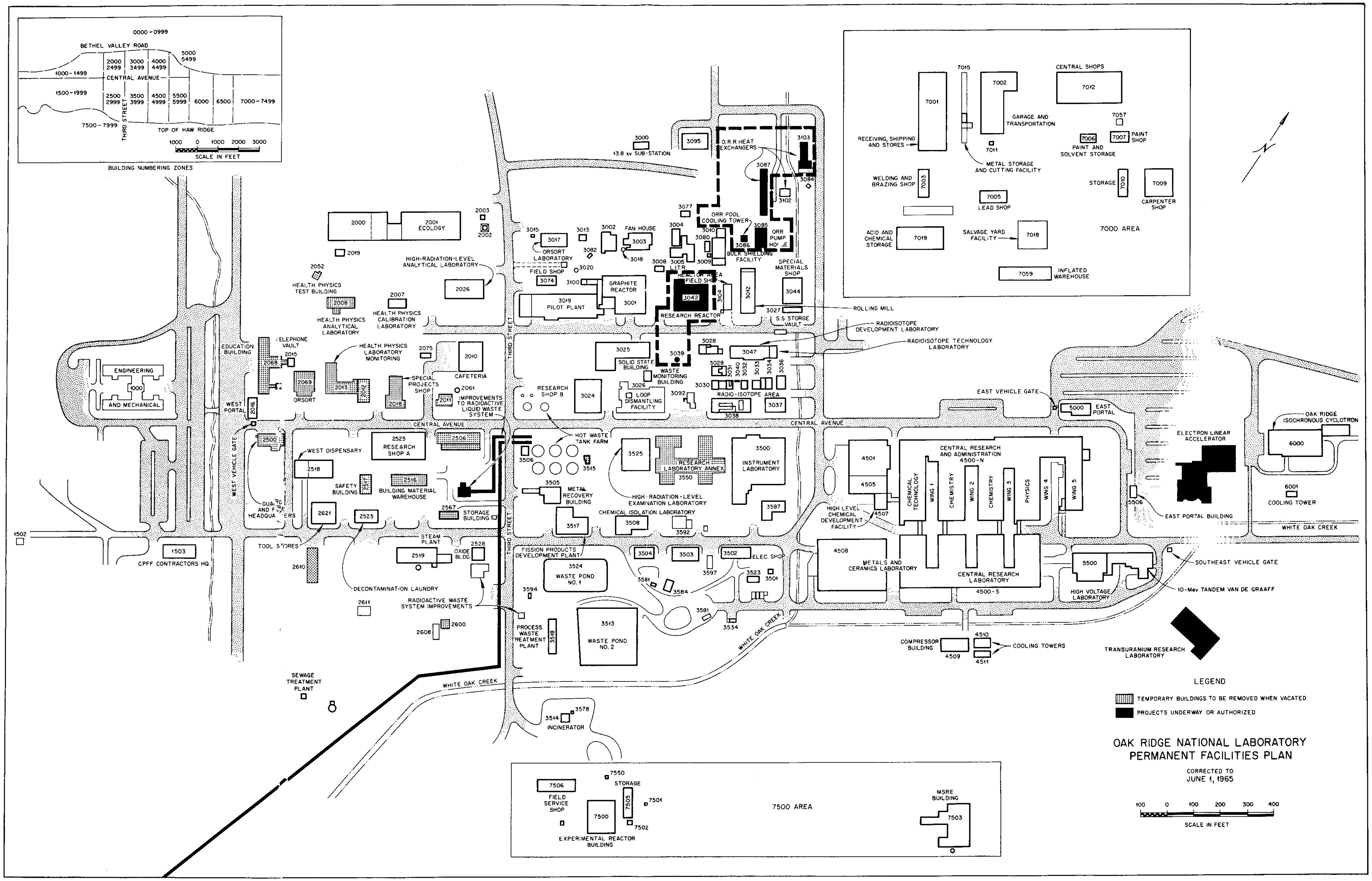

Fig. 2.6. Map of ORNL Facilities.

Table 2.1. Population of the Surrounding Towns[a] Based on 1960 Census

City or Town	Distance from ORR (miles)	Direction	Population	Percent of Time Downwind	
				Night	Day
Oak Ridge	5–12	NNE	27,124	5.6	5.5
Lenoir City	9	SSE	4,979	4.3	6.0
Oliver Springs	9	N	1,163	2.3	2.7
Martel	9	SE	500[b]	1.4	2.8
Coalfield	9	NW	650[b]	0.5	1.1
Windrock	13	N by W	550[b]	2.3	2.7
Kingston	11	WSW	2,010	9.5	11.3
Harriman	13	W	5,931	2.2	3.7
South Harriman	13	W	2,884	2.2	3.7
Petros	14	NW by N	790[b]	1.4	2.8
Fork Mountain	16	NNW	700[b]	2.3	2.7
Emory Gap	15	W	500[b]	2.2	3.7
Friendsville	15	SE	606	1.4	2.8
Clinton	17	NE	4,943	11.6	9.0
South Clinton	17	NE	1,356	11.6	9.0
Powell	17	ENE	500[b]	8.3	6.8
Briceville	20	NNE	1,217	5.6	5.5
Wartburg	20	NW by W	800[b]	1.4	2.8
Alcoa	20	ESE	6,395	2.0	2.0
Maryville	21	ESE	10,348	2.0	2.0
Knoxville	18–25	E	111,827	1.5	2.7
Greenback	20	S by E	960[b]	5.5	4.9
Rockwood	21	W by S	5,343	2.2	3.7
Rockford	22	SE	5,345	1.4	2.8
Fountain City[c]	22	ENE	10,365	8.3	6.8
Lake City	23	NNE	1,914	5.6	5.5
Norris	23	NNE	1,389	5.6	5.5
Sweetwater	23	SSW	4,145	8.4	12.7
Neubert	27	ENE	600[b]	8.3	6.8
John Sevier	27	E	752[b]	1.5	2.7
Madisonville	27	S	1,812	5.5	11.9
Caryville	27	N by E	1,234[b]	9.5	6.1
Sunbright	30	NW	600[b]	0.5	1.1
Jacksboro	30	N by E	679	8.4	12.7

[a]S. E. Beall, R. B. Briggs, and J. H. Westsik, Addendum to ORNL CF-61-2-46, *Molten-Salt Reactor Experiment Preliminary Hazards Report*, pp. 55–56 (Aug. 14, 1961).

[b]1950 census.

[c]Now part of Knoxville.

Table 2.2. Rural Population in Surrounding Counties[a]

County	Total Area[b] (square miles)	Rural Population[c]	Density (number of people per square mile)	Estimated Population		
				Within 10-Mile Radius	Within 20-Mile Radius	Within 30-Mile Radius
Anderson	338	26,600	79	395	14,200	22,800
Blount	584	38,325	66	0	6,720	23,200
Knox	517	138,700	238	13,100	46,400	96,000
Loudon	240	18,800	78	6,080	16,900	18,700
Morgan	539	13,500	25	225	3,625	8,630
Roane	379	12,500	33	3,070	9,170	11,110

[a]S. E. Beall, R. B. Briggs, and J. H. Westsik, Addendum to ORNL CF-61-2-46, *Molten-Salt Reactor Experiment Preliminary Hazards Report*, pp. 55–56 (Aug. 14, 1961).

[b]Does not include area within Oak Ridge reservation.

[c]1960 census – does not include communities with a population of 500 or more.

Table 2.3. Number of Employees in Specific Oak Ridge Areas

Estimated August 1966

Plant or Area	Distance from ORR (miles)	Direction	Number of Employees	
			Group	Total
ORNL				
X-10 area personnel	0–0.50		3400	
Construction personnel	0–1.25		200	
7000 area personnel	1.0	E	300	
Melton Valley personnel	0.75–2.50	SE	110	
Total				4010
ORGDP				
ORGDP area personnel	5.0	W	2500	
Construction personnel	5.0	W	70	
Total				2570
Y-12				
Y-12 area personnel	5.0	NE	5500	
ORNL personnel	5.0	NE	1000	
Construction personnel	5.0	NE	300	
Total				6800
University of Tennessee Agricultural Research Laboratory	5.0	NE		160
Bull Run Steam Plant				
Normal operating personnel (one unit)	11.5	NE	180	
Construction personnel	11.5	NE	620	
Total				800

Table 2.4. Estimated Population Distribution

Radius (Miles)	Sector							
	N	NNE	NE	ENE	E	ESE	SE	SSE
0–0.5	0	0	0	0	0	0	0	0
0.5–1	0	0	0	0	0	0	0	0
1–2	0	0	0	0	0	0	0	0
2–3	0	0	0	0	0	41	20	0
3–4	0	0	0	0	0	87	40	20
4–5	0	0	0	87	87	90	60	40
5–10	7944	20,428	460	7,706	7,706	7,706	783	6546
10–20	5320	13,318	6650	56,414	55,914	23,131	6388	5660

Radius (Miles)	Sector							
	S	SSW	SW	WSW	W	WNW	NW	NNW
0–0.5	0	0	0	0	0	0	0	0
0.5–1	0	0	0	0	0	0	0	0
1–2	0	0	0	0	0	0	0	0
2–3	90	90	90	45	0	0	0	0
3–4	135	135	135	45	0	0	0	0
4–5	180	180	180	45	2,751	0	0	200
5–10	1567	1564	781	781	781	781	781	781
10–20	4700	4700	1563	3573	16,741	2542	1500	4190

2.3 Geophysical Features

2.3.1 Meteorology

Oak Ridge is located in a broad valley between the Cumberland Mountains, which lie to the northwest of the area, and the Great Smoky Mountains, which lie to the southeast. These mountain ranges are oriented northeast to southwest. The valley between them is corrugated by broken ridges 300 to 500 ft high oriented parallel to the main valley. The local climate is noticeably influenced by topography.

Temperature.[1] – The coldest month is normally January, but differences between the mean temperatures of the three winter months of December, January, and February are comparatively small. July is usually the hottest month, but differences between the mean temperatures of the summer months of June, July, and August are also comparatively small. Mean temperatures of the spring and fall months progress in orderly fashion from cooler to warmer and warmer to cooler, respectively, without a secondary maximum or minimum. Temperatures of 100°F or higher are unusual, having occurred during less than one-half of the years of the period on record, and temperatures of zero and below are rare.

The annual mean maximum and minimum temperatures are 69.4 and 47.6°F, respectively, with an annual mean temperature of 58.5°F. The extreme low and high temperatures are −10°F and +103°F, recorded in January 1966 and September 1954 respectively. Table 2.5 lists the average monthly temperature range based on the period 1931 to 1960, adjusted to represent observations taken at the present standard location of the weather station.

Vertical Temperature Gradient. – Information on the temperature gradient and mean wind speed for

[1]U.S. Department of Commerce, Weather Bureau, Local Climatological Data with Comparative Data 1962, Oak Ridge, Tennessee, Area Station (X-10).

Table 2.5. ORNL Climatological Standard Normal Temperatures (1931–1960)

Month	Temperature (°F)		
	Maximum	Minimum	Average
January	48.9	31.2	40.1
February	51.6	31.8	41.7
March	58.9	37.0	48.0
April	70.0	46.3	58.2
May	79.0	54.8	66.9
June	86.1	63.3	74.7
July	88.0	66.7	77.4
August	87.4	65.6	76.5
September	83.0	59.2	71.1
October	72.2	47.7	60.0
November	58.6	36.5	47.6
December	49.4	31.3	40.4
Annual	69.4	47.6	58.5

each month is found in a recent report on meteorology of the Oak Ridge area.[2] The seasonal and annual averages as derived from this information are presented in Fig. 2.7.

Precipitation.[3] – Precipitation in the ORNL area is normally well distributed throughout the year, with the drier part of the year occurring in the early fall. Winter and early spring are the seasons of heaviest precipitation, with the monthly maximum normally occurring from January to March. A secondary maximum, due to afternoon and evening thundershowers, occurs in the month of July. September and October are usually the driest months.

The average and maximum annual precipitations are 51.52 and 66.2 in. respectively. The maximum rainfall in the area in a 24-hr period was 7.75 in., recorded in September 1944. The recurrence interval of this amount of precipitation in a 24-hr period has been estimated to be about 70 years. The greatest average monthly precipitation normally occurs in March and has a value of 5.44 in.

The average monthly precipitation is given in Table 2.6.

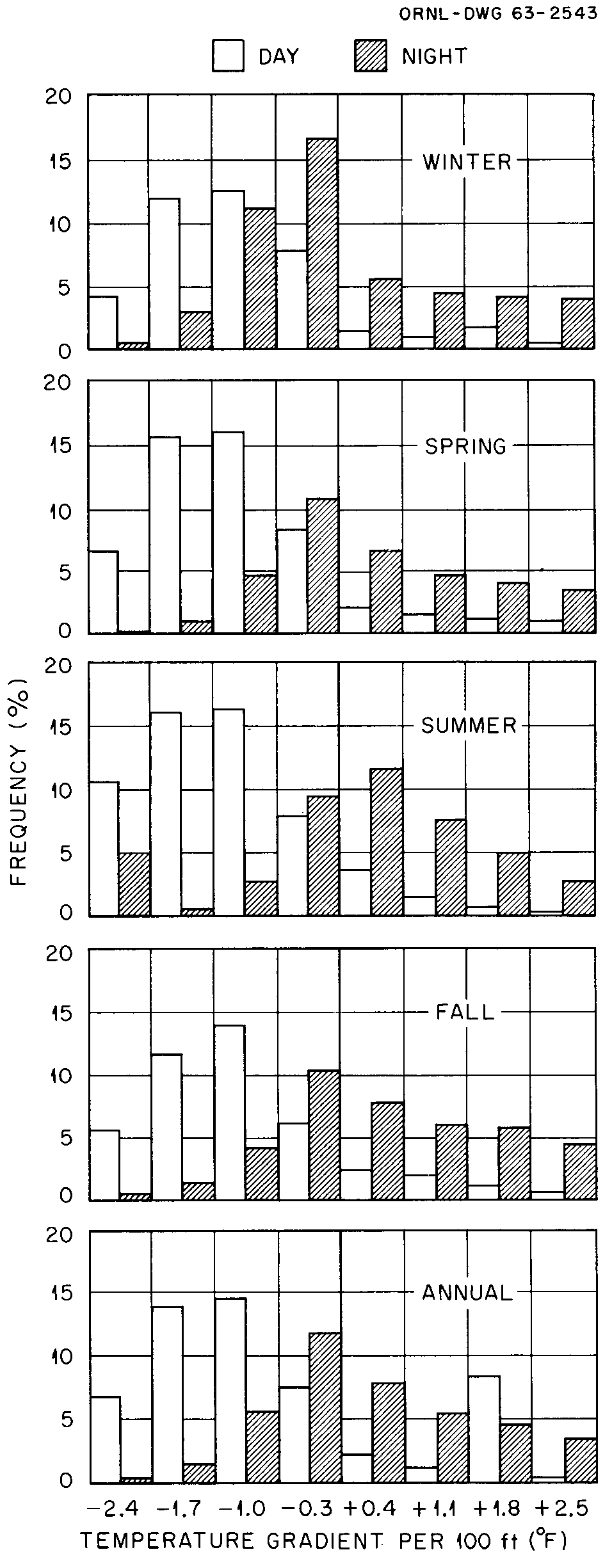

Fig. 2.7. Vertical Temperature Gradient.

[2]W. F. Hilsmeier, *Supplementary Meteorological Data for Oak Ridge*, ORO-199 (Mar. 15, 1963).

[3]U.S. Department of Commerce, Weather Bureau, Local Climatological Data with Comparative Data 1962, Oak Ridge, Tennessee, Area Station (X-10).

Table 2.6. ORNL Average Monthly Precipitation Data

Month	Precipitation (in.)
January	5.24
February	5.39
March	5.44
April	4.14
May	3.48
June	3.38
July	5.31
August	4.02
September	3.59
October	2.82
November	3.49
December	5.22
Total	51.52

Light snow usually occurs in all the months from November to March, but the total monthly snowfall is often only a trace. The total snowfall for some winters is less than 1 in. The average annual snowfall for the period from 1948 to 1961 was 6.9 in. The maximum snowfall in a 24-hr period was 12.0 in., which occurred in March 1960. The largest monthly snowfall (21.0 in.) also occurred in March 1960.

The heavy fogs that occasionally occur are almost always in the early morning and are of relatively short duration.

Wind.[4] – The valleys in the vicinity of the ORNL site are oriented northeast to southwest, and considerable channeling of the winds in the valley occurs. This is evident in Fig. 2.8, which shows the annual frequency distribution of winds in the vicinity of ORNL. The flags on these wind-rose diagrams point in the direction from which the wind comes. The prevailing wind directions are up-valley from southwest and west-southwest approximately 40% of the time, with a secondary maximum of down-valley winds from northeast and east-northeast 30% of the time. The prevailing wind regimes reflect the orientation of the broad valley between the Cumberland Plateau and the Smoky Mountains, as well as the orientation of the local ridges and valleys. The gradient wind in this latitude is usually southwest or westerly, so the daytime winds tend to reflect a mixing of the gradient winds. The night winds are the results of drainage of cold air down the local slopes and the broader Tennessee Valley. The combination of these two effects, as well as the daily changes in the pressure patterns over this area, gives the elongated shape to the typical wind roses.

During inversions, the northeast and east-northeast winds occur most frequently, usually at the expense of the southwest and west-southwest winds. The predominance of light northeast and east-northeast winds under stable conditions is particularly noticeable in the summer and fall, when the lower wind speeds aloft and the smaller amount of cloudiness allow the nocturnal drainage patterns to develop.

Wind roses prepared from five years (1956 to 1960) of data[2] are shown in Figs. 2.9 and 2.10. These represent the wind direction, its frequency, and the percent calm under inversion and lapse conditions in the ORNL area.

Considerable variation is observed in both wind speed and direction within small areas in Bethel Valley. In general, at night or under stable conditions, the winds tend to be from the northeast or east-northeast and rather light in the valleys, regardless of the gradient wind. However, strong winds aloft will control the velocities and directions of the valley winds, reversing them or producing calms when opposing the local drainage. During the day, the surface winds tend to be in the same direction as winds aloft, with increasing conformity as the upper wind speed increases. Only with strong winds aloft or winds parallel to the valleys would it be of value to attempt to extrapolate air movements for any number of miles by using valley winds. In a well-developed stable situation, however, a very light air movement will follow a valley as far as the valley retains its structure, even though the prevailing winds a few hundred feet above the ground are in an entirely different direction. In any particular valley location, the wind direction will be governed by the local valley wind regime and the degree of coupling with the upper winds.[4]

A comparison of pilot balloon observations made throughout 1949 and 1950 at Knoxville and Oak Ridge shows that above about 2000 ft the wind roses at these two stations are almost identical.

[4]*Aircraft Reactor Experiment Hazards Summary Report*, ORNL-1407 (Nov. 24, 1952).

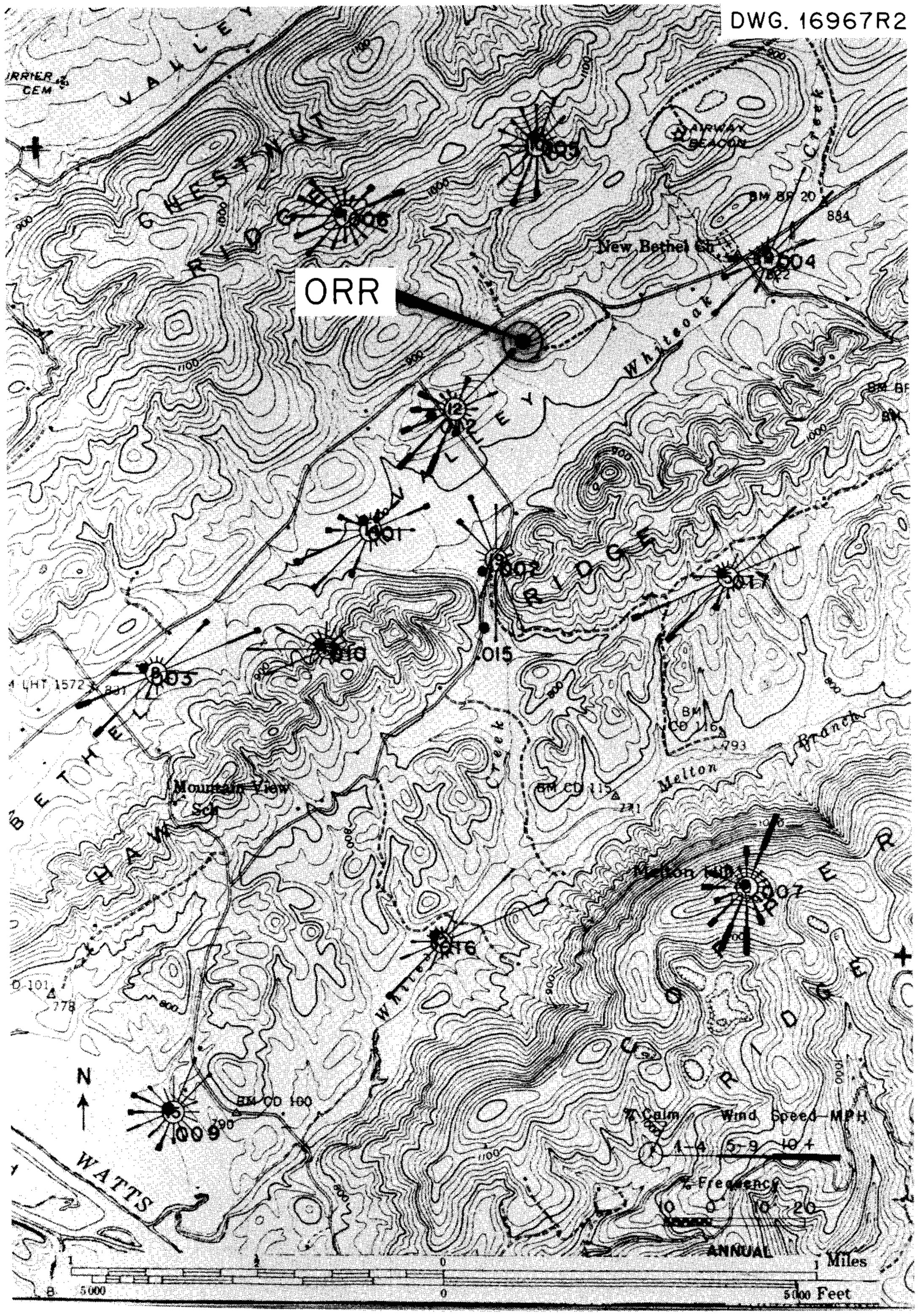

Fig. 2.8. Annual Frequency Distribution of Winds in the Vicinity of ORNL.

This similarity of data makes possible the use of the longer period of record (1927 to 1950) for Knoxville and tends to minimize the importance of abnormalities introduced by the use of the short record at Oak Ridge.

Annual wind roses are shown for Knoxville (1927 to 1950) in Fig. 2.11. Since pilot balloon observations are made only when low clouds, dense fog, and precipitation are absent, they are not representative of the upper wind at all times.

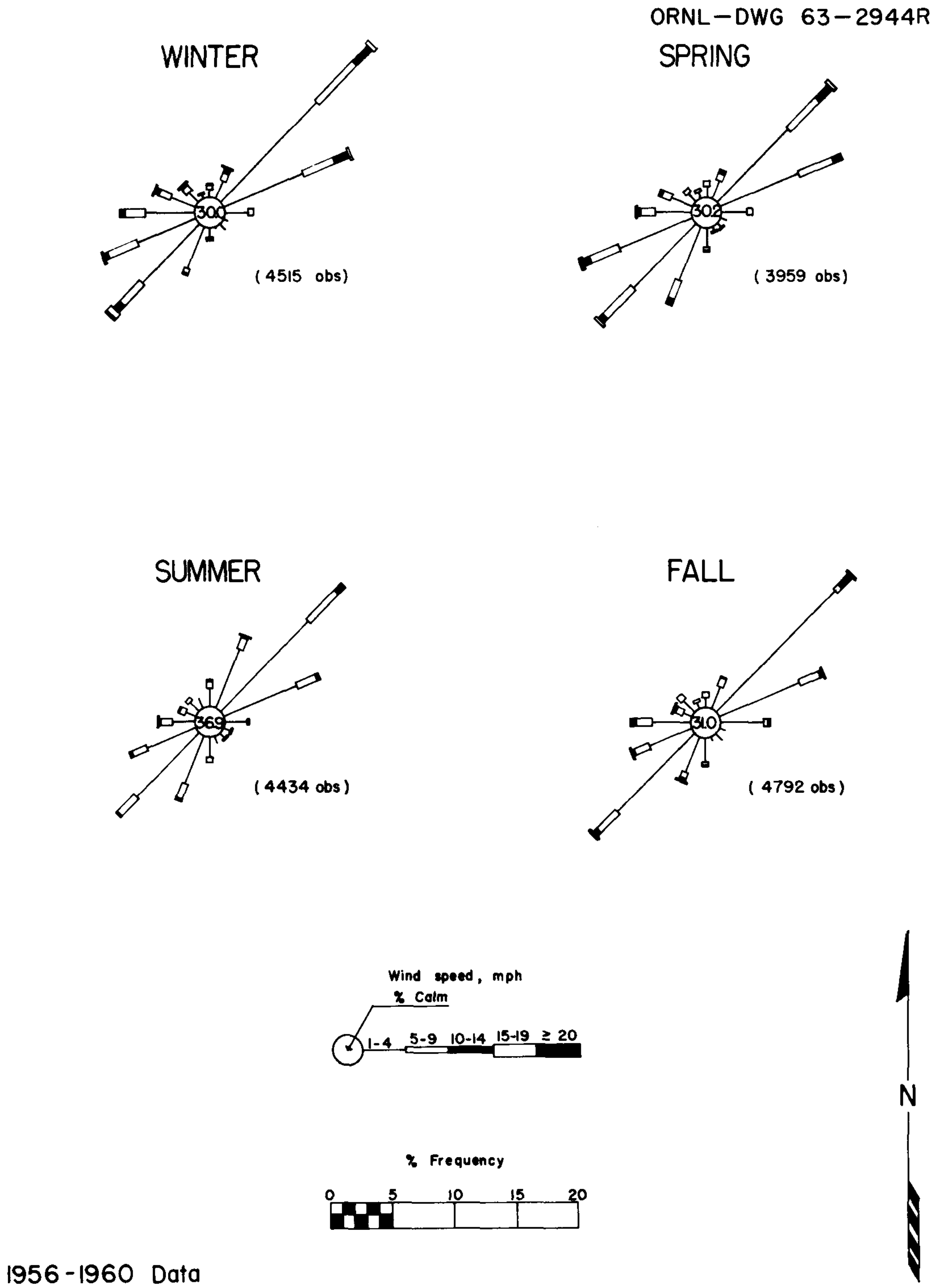

Fig. 2.9. ORNL Area Seasonal Wind Roses, Inversion Conditions.

Three years of radio wind-balloon data for Nashville (1947 to 1950) are available. These are observations taken without regard to the current weather at the time of observation. Comparison of these wind roses for Knoxville and Nashville indicates that the modes for winds above 3000 m above mean sea level should be shifted to west instead of west-northwest when observations with rain are included in the set.

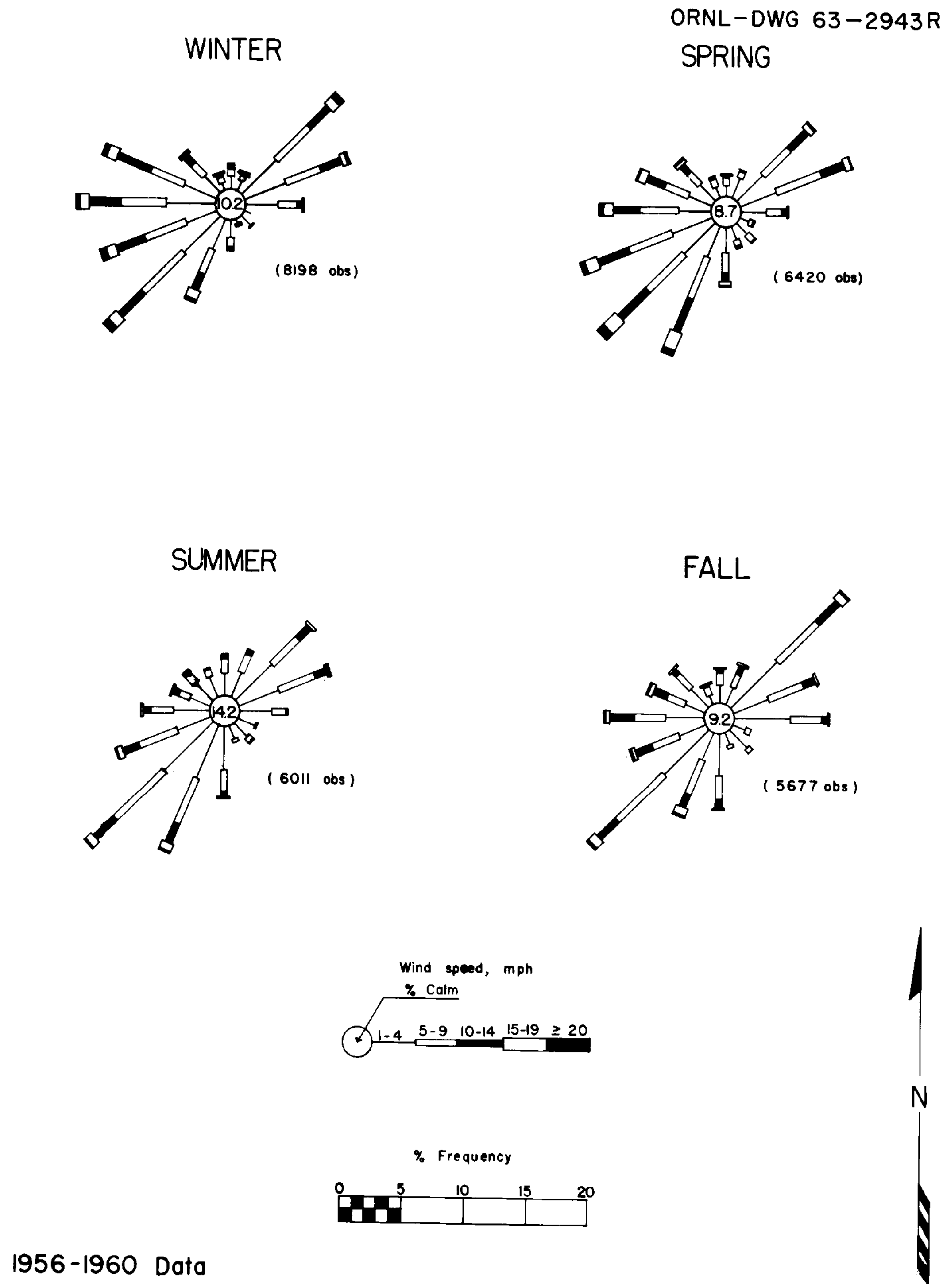

Fig. 2.10. ORNL Area Seasonal Wind Roses, Lapse Conditions.

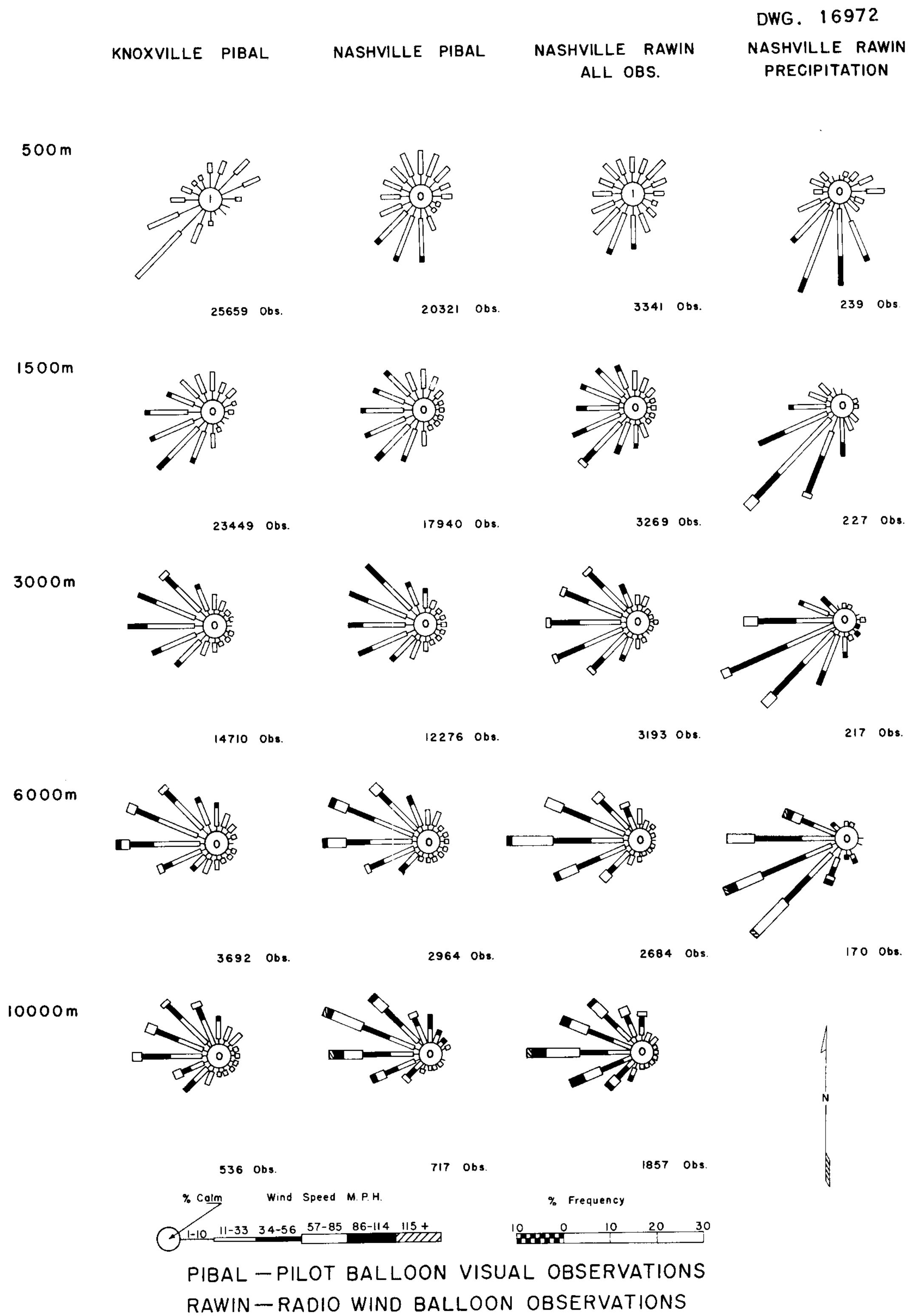

Fig. 2.11. Wind Roses at Knoxville and Nashville for Various Altitudes.

Table 2.7. Atmospheric Stability Constants at Oak Ridge[a]

Condition	Occurrence (%)
A	Never
B	8
C	40
D	20
E	22
F	10

[a]W. F. Hilsmeier, AEC-ORO, to T. H. Row, ORNL, private communication, June 27, 1963.

Table 2.8. Summary of Seasonal Frequency of Inversions

Season	Frequency of Inversion (%)
Winter	31.8
Spring	35.1
Summer	35.1
Fall	42.5
Annual	35.9

The northeast to southwest orientation of the valley between the Cumberland Plateau and the Smoky Mountains influences the wind distribution over the Tennessee Valley up to an altitude of about 5000 ft, although the variations in the Valley do not extend above about 2000 ft. Above about 5000 ft, the southwesterly mode is dominated by the prevailing westerlies usually observed at these latitudes.

Previous investigation of the relation of wind direction to precipitation indicated that the distribution in wind directions is little different from that of wind observations made without precipitation.[4] This is consistent with the experiences of forecasters to the extent that there is little correlation between surface wind direction and rain, particularly in rugged terrain. Figure 2.11 shows the upper winds measured at Nashville during the period 1948 to 1950 when precipitation was occurring at observation time. In general, the prevailing wind at a given level is shifted to the southwest from west and to the south-southwest from the southwest, with the shift being most marked in the winter. Wind velocities are somewhat higher during the occurrence of precipitation.

Tornadoes rarely occur in the valley between the Cumberlands and the Great Smokies, and it is highly improbable that winds stronger than 100 mph would ever occur at the ORNL site.

Atmospheric Diffusion Characteristics. – The method proposed by Pasquill[5] is widely used to calculate the dispersion of airborne materials in the lower atmosphere. This method requires a knowledge of two parameters known as the horizontal and vertical dispersion coefficients. These coefficients have been tabulated according to the stability of the lower atmosphere for six different conditions, identified as A to F. Condition A represents extreme instability, and condition F represents extremely stable conditions. Table 2.7 gives the frequency of these various conditions at the ORNL site.

A study[6] of the average duration of inversions in the Oak Ridge area indicates that they have a length of 8 hr during winter, 9 hr during spring, 9 hr during summer, and 10 hr during fall.

The percentage of time during which inversion conditions exist is given for the four seasons in Table 2.8.

Atmospheric contamination by long-lived fission products and fallout occurring in the general environment of the Oak Ridge area is monitored by a number of stations surrounding the area.[7] This system provides data to aid in evaluating local conditions and to assist in determining the spread or dispersal of contamination should a major incident occur.

Environmental Radioactivity. – Data on the environmental levels of radioactivity in the Oak Ridge area[7] are given in Tables 2.9 through 2.12.

[5]F. Pasquill, "The Estimation of the Dispersion of Windborne Material," *Meteorol. Mag.* 90(1063), 33 (1961).

[6]W. M. Culkowski, AEC-ORO, to T. H. Row, ORNL, private communication, July 8, 1963.

[7]*Applied Health Physics Annual Report for 1962*, ORNL-3490 (Sept. 25, 1963).

Table 2.9. Concentration of Radioactive Materials in Air – 1962

Averaged weekly from filter-paper data

Station No.	Location	Long-Lived Activity ($\mu c/cm^3$)	Number of Particles by Activity Ranges[a] $<10^5$ dis/24 hr	10^5–10^6 dis/24 hr	10^6–10^7 dis/24 hr	$>10^7$ dis/24 hr	Total	Particles per 1000 ft^3
			Laboratory Area					
		$\times 10^{-13}$						
HP-1	S 3587	38	128	1.6	0.00	0.00	129	3.1
HP-2	NE 3025	43	122	1.9	0.04	0.00	124	3.5
HP-3	SW 1000	37	129	2.1	0.10	0.02	131	2.1
HP-4	W Settling Basin	21	91	1.2	0.04	0.00	93	1.6
HP-5	E 2506	51	115	1.2	0.04	0.04	117	3.9
HP-6	SW 3027	33	136	1.5	0.02	0.02	137	2.4
HP-7	W 7001	40	115	1.8	0.00	0.00	117	2.3
HP-8	Rock Quarry	39	132	1.5	0.00	0.02	133	2.5
HP-9	N Bethel Valley Rd.	31	145	1.6	0.00	0.00	146	2.3
HP-10	W 2075	38	126	1.3	0.00	0.00	128	3.1
Average		37	124	1.6	0.02	0.01	125	2.7
			Perimeter Area					
HP-31	Kerr Hollow Gate	34	135	1.6	0.04	0.04	137	2.7
HP-32	Midway Gate	37	132	2.1	0.02	0.00	134	2.6
HP-33	Gallaher Gate	32	113	1.4	0.00	0.02	114	2.2
HP-34	White Wing Gate	34	153	1.5	0.00	0.00	155	3.0
HP-35	Blair Gate	39	168	1.6	0.00	0.02	169	3.3
HP-36	Turnpike Gate	39	158	2.2	0.02	0.04	161	3.2
HP-37	Hickory Creek Bend	34	114	1.6	0.02	0.00	115	2.3
Average		36	139	1.7	0.01	0.02	141	2.8
			Remote Area					
HP-51	Norris Dam	43	139	2.3	0.04	0.00	141	2.6
HP-52	Loudon Dam	42	130	2.8	0.10	0.00	133	2.4
HP-53	Douglas Dam	44	150	2.6	0.02	0.00	153	2.8
HP-54	Cherokee Dam	40	164	2.4	0.04	0.02	167	3.0
HP-55	Watts Bar Dam	45	157	2.0	0.04	0.00	159	2.9
HP-56	Great Falls Dam	46	166	2.3	0.00	0.00	168	3.1
HP-57	Dale Hollow Dam	38	171	1.6	0.00	0.04	172	2.9
Average		43	154	2.3	0.03	0.01	157	2.8

[a]Determined by filtration techniques.

Table 2.10. Radioparticulate Fallout – 1962

Averaged weekly from gummed paper data

Station No.	Location	Long-Lived Activity (μc/cc)	Number of Particles by Activity Ranges: $<10^5$ dis/24 hr	10^5–10^6 dis/24 hr	10^6–10^7 dis/24 hr	$>10^7$ dis/24 hr	Total	Total Particles per ft^2
			Laboratory Area					
		$\times 10^{-13}$						
HP-1	S 3587	15	79	2.1	0.12	0.06	81	42
HP-2	NE 3025	17	88	2.3	0.04	0.06	91	49
HP-3	SW 1000	15	83	2.0	0.15	0.06	86	42
HP-4	W Settling Basin	14	73	2.3	0.08	0.04	75	48
HP-5	E 2506	14	86	2.0	0.08	0.04	91	50
HP-6	SW 3027	16	101	2.9	0.02	0.02	104	61
HP-7	W 7001	15	89	2.4	0.02	0.06	92	49
HP-8	Rock Quarry	17	89	2.6	0.00	0.08	91	46
HP-9	N Bethel Valley Rd.	16	88	2.9	0.06	0.12	91	41
HP-10	W 2075	15	100	2.3	0.04	0.00	103	55
Average		15	88	2.4	0.06	0.05	91	48
			Perimeter Area					
HP-31	Kerr Hollow Gate	17	103	2.13	0.13	0.10	105	47
HP-32	Midway Gate	16	99	2.6	0.10	0.06	102	46
HP-33	Gallaher Gate	14	82	2.4	0.10	0.00	85	42
HP-34	White Wing Gate	18	104	2.2	0.19	0.08	106	47
HP-35	Blair Gate	15	124	2.0	0.06	0.04	126	50
HP-36	Turnpike Gate	16	109	3.5	0.08	0.02	112	57
HP-37	Hickory Creek Bend	16	85	2.3	0.04	0.08	87	47
Average		16	101	2.5	0.10	0.05	103	48
			Remote Area					
HP-51	Norris Dam	14	86	2.2	0.12	0.04	89	36
HP-52	Loudon Dam	13	70	2.7	0.06	0.06	73	29
HP-53	Douglas Dam	13	77	2.7	0.06	0.08	80	35
HP-54	Cherokee Dam	14	81	2.9	0.13	0.06	84	35
HP-55	Watts Bar Dam	16	81	2.2	0.14	0.08	83	37
HP-56	Great Falls Dam	14	98	2.2	0.06	0.02	100	39
HP-57	Dale Hollow Dam	14	96	2.0	0.08	0.06	98	33
Average		14	84	2.4	0.09	0.06	87	35

Table 2.11. Concentration of Radioactive Materials in Rainwater – 1962

Averaged weekly by stations

Station No.	Location	Activity in Collected Rainwater ($\mu c/cm^3$)
	Laboratory Area	
		$\times 10^{-7}$
HP-7	W 7001	10.3
	Perimeter Area	
HP-31	Kerr Hollow Gate	11
HP-32	Midway Gate	12
HP-33	Gallaher Gate	10
HP-34	White Wing Gate	11
HP-35	Blair Gate	11
HP-36	Turnpike Gate	10
HP-37	Hickory Creek Bend	11
Average		11
	Remote Area	
HP-51	Norris Dam	14
HP-52	Loudon Dam	11
HP-53	Douglas Dam	13
HP-54	Cherokee Dam	11
HP-55	Watts Bar Dam	14
HP-56	Great Falls Dam	16
HP-57	Dale Hollow Dam	11
Average		13

Table 2.12. Radioactive Content of Clinch River – 1962

Location	Concentration of Nuclides of Primary Concern in Units of 10^{-8} $\mu c/cm^3$						Average Concentration of Total Radioactivity 10^{-8} $\mu c/cm^3$	$(MPC)_w$[a] 10^{-6} $\mu c/cm^3$	Percent of $(MPC)_w$
	Sr^{90}	Ce^{144}	Cs^{137}	$Ru^{103-106}$	Co^{60}	Zr^{95}–Nb^{95}			
Mile 41.5	0.16	0.14	0.02	0.78	*b*	0.42	1.5	0.90	1.7
Mile 20.8[c]	0.15	0.02	0.09	21	0.18	0.09	34	4.6	7.4
Mile 4.5[d]	0.34	0.20	0.07	16	0.32	0.54	17	3.5	4.9

[a]Weighted average calculated for the mixture, using $(MPC)_w$ values for specific radionuclides recommended in *NBS Handbook* 69.

[b]None detected.

[c]Values given for this location are calculated values based on the levels of waste released and the dilution afforded by the river.

[d]Center's Ferry (near Kingston, Tenn., just above entry of the Emory River).

2.3.2 Regional Topography and Geology

The area under consideration is within the Oak Ridge Reservation in Roane and Anderson Counties, Tennessee. White Oak Creek is a tributary of the Clinch River, entering that river from the north bank just above Jones Island at about mile 20.8 of the Clinch. The watershed of the creek, which extends in a generally northeast direction from its mouth, has an area of 6 sq miles and is roughly diamond shaped. It lies primarily in Roane County, with a very small portion of the upper watershed in Anderson County.

The topography is typical of the Valley and Ridge area, characterized by ridges and valleys running northeast to southwest, which are a result of the geological structure and stratigraphy. The rocks in the area dip 20 to 30° in approximately a southeasterly direction. The more resistant rocks support the ridges, and the less resistant ones have been eroded to form the valleys. The same formations are repeated by the intervention of a major thrust fault. Elevations in the area range from about 750 ft where White Oak Creek enters the Clinch River to 1356 ft at Melton Hill, giving a maximum relief of about 600 ft. The ground elevation at the ORR is about 820 ft.

Four principal rock formations are present in the area: the Rome sandstone, of Cambrian age; the Conasauga shale, also of Cambrian age; the Chicamauga limestone, of Ordovician age; and the Knox dolomite, of Cambro-Ordovician age.

The Rome formation, which forms Haw Ridge, consists of evenly bedded, fine-grained sandstone and shale of red, green, and other colors. In the Oak Ridge area the Rome formation is more than 1000 ft thick.

Layers of the Rome formation dip beneath the Conasauga shale, which underlies all of Melton Valley. Although the residual material that covers the Conasauga is quite uniform in appearance, the formation may be subdivided into four distinct types of rocks on the basis of core drilling. At its base is a zone of dark-red silty shale about 300 ft thick with numerous thin beds of light-green sandstone. Overlying this layer is about 450 ft of dark-gray calcareous shale with numerous thin beds of light-gray crystalline limestone. Above this is a transitional zone of interbedded shale and limestone. The top zone of the Conasauga, which forms the northwest slope of Copper Ridge, is predominantly limestone. In Bear Creek Valley this zone is about 300 ft thick.

In Bethel Valley, the Knox group of formations, about 2600 ft thick, lies above the Conasauga. It consists largely of light- to dark-gray cherty dolomitic limestone. It is a ridge former and underlies Haw Ridge, immediately southeast of Bethel Valley, and Chestnut Ridge, immediately northwest of Bethel Valley. The Knox is nearly everywhere covered by a cherty residual white to red clay soil that ranges from 30 to more than 100 ft in thickness.

The Chicamauga limestone, which stratigraphically overlies the Knox, underlies Bethel Valley, in which the X-10 plant is located. This unit, which is about 1000 ft thick in the area, consists of thinly bedded limestone and shale. Bedrock is overlain by a thin blanket of residual clay soil rarely more than 10 ft thick.

2.3.3 Regional Hydrology

Groundwater. – Information on the occurrence of groundwater in the sandstone and shale of the Rome formation is sparse, since few wells have been drilled in the formation. However, the Rome is well expressed in many road cuts through the water gaps in Haw Ridge; observations of its physical properties indicate that the formation is probably quite impermeable and that the rate of groundwater movement through it is relatively low.

The Chicamauga limestone formations in Bethel Valley are almost devoid of permeability below a depth of about 100 ft. Exploratory drilling in Bear Creek Valley, which is underlain by the Conasauga group, indicated that limestones in the upper part of the Conasauga contain cavities several feet wide and extending at least 100 ft below the surface.

In general, the ability of the Conasauga to transmit water or other fluids increases with increasing lime content. Hence the lower two members, which underlie the northwest side of Bethel Valley, transmit water less readily than the more limy upper members, which underlie the valley to the southeast. Observations of water levels in wells in Bethel Valley indicate that the water table may be as much as 60 ft below the land surface on upland areas, but at points of lower elevation along White Oak Creek it is at or just beneath the ground surface.

Groundwater in the Knox formation occurs in, and moves through, solution cavities, some of which are cavernous. There are many sinkholes in the

Table 2.13. Stream Flow at Various Stations on White Oak Creek and Melton Branch

Drainage Area (sq miles)	Records Available	Number of Years	Average Discharge (cfs)	Maximum Discharge		Minimum Daily Discharge	
				Cfs	Date	Cfs	Date
			White Oak Creek at White Oak Lake Dam				
6.0	7/1953–9/1955	2	11.2	669	12/29/54	No flow	July 17–20, 1953; Dec. 1, May 18–19, 1954; Sept. 3–6, 1955
			White Oak Creek Below ORNL[a]				
3.6	6/1950–7/1953, 7/1955 to date	6	9.37	642	8/30/50	1.9	Oct. 2, 1950
			Melton Branch, Upstream from White Oak Lake[b]				
1.5	8/1955 to date	4	2.228	121	1/21/59	No flow	Many days during August through November 1955
			White Oak Creek at ORNL[c]				
2.1	6/1950–7/1955			616	8/2/50	0.7	Nov. 2, 1950; Aug. 2, 8, 12, 13, 21, 26, 28, 30, and 31, 1951; Sept. 8 and 9, 1951; Oct. 14, 1951; July 25 and 26, 1953

[a]0.1 mile upstream from mouth of Melton Branch.
[b]0.1 mile upstream from White Oak Creek.
[c]1.2 miles upstream from Melton Branch, 1000 ft above effluent from settling basins.

areas of outcrop, and sizable springs issue from the bases of the ridges. Because of the high permeability of the cavernous bedrock, which permits free and rapid movement of groundwater, the depth of water may exceed 100 ft along ridge tops. Frequently the position of the water table coincides approximately with the interface between rock and residual clay overburden.

Stream Flow. – The U.S. Geological Survey has operated gaging stations to obtain continuous records of stream flow at sections on White Oak Creek and Melton Branch. The information from four stations is summarized in Table 2.13; the natural flow at all four is affected by operations at ORNL.

2.3.4 ORR Site Topography, Geology, and Hydrology

In 1948–49 an investigation was made of the geology and hydrology of the X-10 area. The following description is based principally on information in the report[8] of this investigation.

Topography and Geology. – Oak Ridge National Laboratory is located in Bethel Valley and covers an area about ½ mile long and ½ mile wide which comprises about 160 acres. Elevations in the

[8]P. B. Stockdale, *Geologic Conditions at the Oak Ridge National Laboratory X-10 Area Relevant to the Disposal of Radioactive Waste*, ORO-58 (Aug. 1, 1951).

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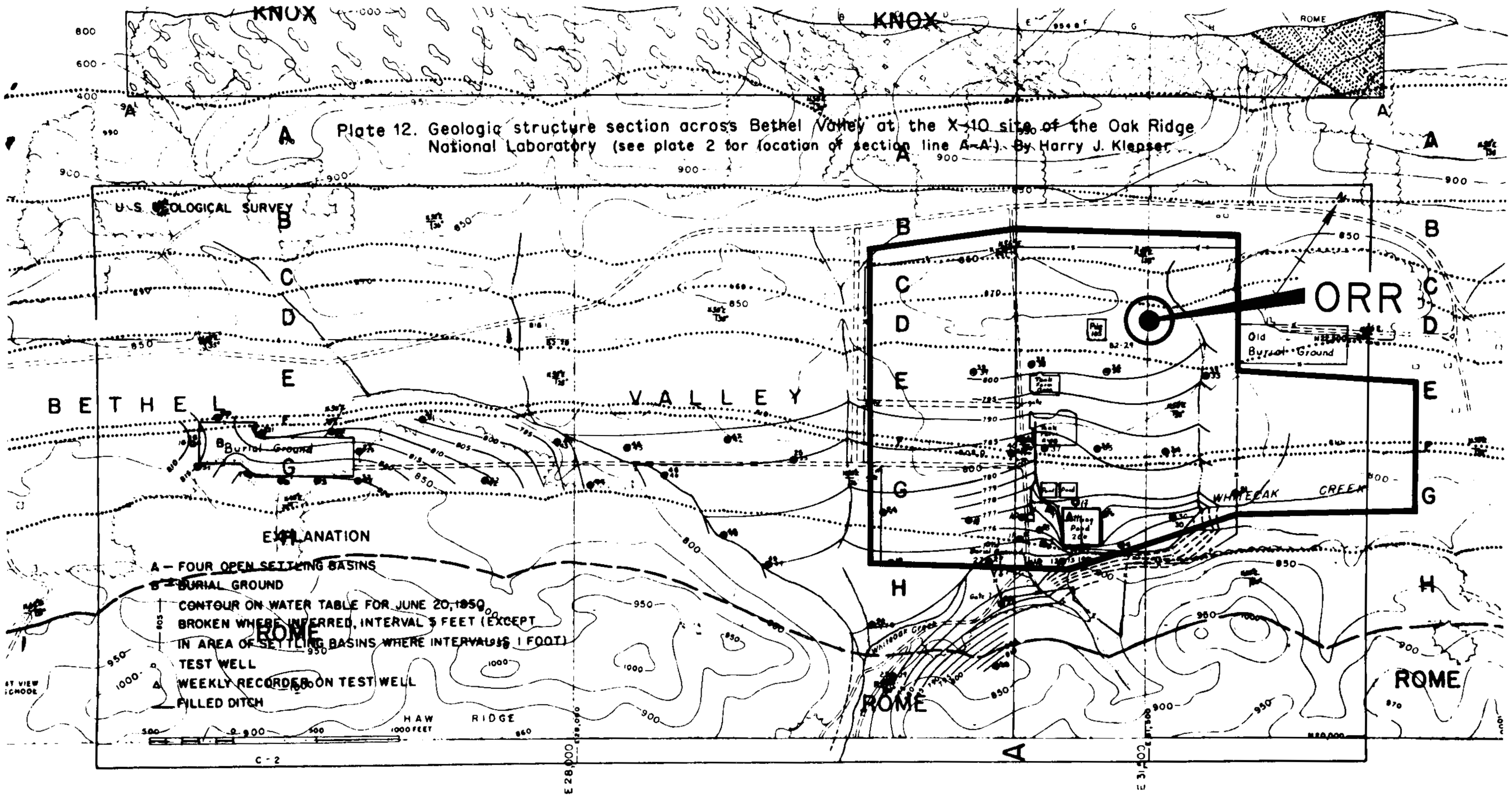

Fig. 2.12. Geologic and Water Table Map of ORNL.

plant area range from 780 to 900 ft above mean sea level. The Laboratory is bounded on the northwest by Chestnut Ridge, with elevations up to 1200 ft, and on the southeast by Haw Ridge, with elevations up to 1000 ft.

The entire plant area is underlain by rocks of the Chicamauga group, of Ordovician age. In this area the thickness of the group is 1735 ft. The rocks are mainly limestone; however, variations in the types of rocks permit their separation into eight distinguishable and mappable subdivisions (Fig. 2.12).

The composite section of the Chicamauga group, in descending order of units, is summarized in Table 2.14.

The direction of dip of all the rocks in the X-10 area is toward the southeast, and the angle of dip is between 30 and 40°. The average direction of their strike is north 58° east.

The Chicamauga group is covered by a mantle of clayey soil derived from the decomposition of the underlying consolidated bedrock. The unconsolidated, weathered material ranges in thickness from about 1 to 25 ft, averaging perhaps 10 ft.

Hydrology. – The area around the ORR is drained by White Oak Creek and its tributaries. Just south of the Laboratory, White Oak Creek flows out of Bethel Valley through a gap in Haw Ridge. Information on the discharge of White Oak Creek is given in Sect. 2.3.3.

Groundwater in the rock beneath the ORR is derived from precipitation that falls on the area and its immediate surroundings, and it is constantly moving from points of recharge to points of discharge at lower elevations. Thus groundwater discharge contributes to the base flow of the surface streams in the area and ultimately augments the flow of the Clinch River. Flow rates of the rivers fed by or affected by this flow are indicated in Table 2.15.

The limestones and shales of the Chicamauga group are quite impermeable and are incapable of rapidly transmitting large amounts of water. The openings in the rocks are confined to narrow seams developed along bedding planes or joints, which decrease in size with depth. Below about 100 ft the rock is generally devoid of openings, so it is probable that most of the groundwater in the area moves through the unconsolidated soils near the surface rather than the rock.

The depth to water in the ORR area ranges from 25 to 30 ft below land surface at points of high elevations to land surface or to near land surface along the surface streams. Figure 2.12 shows the configuration of the water table on June 20, 1950.

Table 2.16 lists the community water systems in Tennessee downstream from ORNL which are supplied by intakes on the Clinch or Tennessee Rivers or their tributaries.

Table 2.14. Composition of Chicamauga Formation at the ORR Site

Unit	Description of Rock	Thickness (ft)	
H	Siltstone, calcareous, gray, olive maroon; with shaly partings and thin limestone lenses	85	
	Limestone of varied types; gray, olive-gray buff, drab; mostly thin-bedded; with argillaceous partings; weathers to shaly appearance; with fossiliferous zones	180	
	Limestone, argillaceous (calcareous siltstone), gray, olive gray, pinkish maroon; even-bedded; with shale partings	35	300
G	Limestone of various types, dark gray to brownish gray; mostly nodular with abundant black irregular clay partings; dense to medium grained; mostly thin-bedded, partly massive; with shale partings; weathers to a lighter-colored shaly or nodular appearance; with some fossiliferous horizons; mostly covered in lowlands		300

Table 2.14 (continued)

Unit	Description of Rock	Thickness (ft)	
F	Siltstone, calcareous, alternating with shale; olive gray to maroon; even-bedded; laminated; weathers to a red shaly appearance; produces a slight rise in topography; a very distinctive unit		25
E	Limestone, mostly gray to drab, partly pinkish maroon, mottled; brittle, thin-bedded to massive; with shaly partings	60	
	Limestone, similar to unit G, mostly covered in lowlands	220	
	Calcareous shale and argillaceous limestone, gray to buff; in alternating thin even beds; yielding small roundish slabs upon weathering, with yellow-buff color	45	
	Limestone of various types, gray; most argillaceous and nodular; in thin irregular beds with shale partings; abundant fossils	55	
			380
D	Limestone and chert; limestone is gray to olive gray, in part nodular, shaly, and thin-bedded, in part massive; with abundant chert in thin, even bands, breaking into angular fragments upon weathering; produces a chain of low hills		160
C	Shale, calcareous, olive gray to light maroon; fossiliferous; evenly laminated	10	
	Limestone of various types, gray; fine- to coarse-grained, partly crystalline, partly nodular; mostly massive; with occasional patches of chert; partly fossiliferous; "quarry beds"	105	
			115
B	Siltstone, in even beds up to 2 ft thick, laminated, alternating with calcareous shale; olive gray, buff, maroon; some limestone, nonresistant; more shale at base		215
A	Limestone of various types, dark gray to buff; with shale partings; with gray to black chert in nodules and lenses	80	
	Chert, thin-bedded, with shaly partings	15	
	Siltstone, calcareous, olive gray to maroon; weathers to shaly appearance	30	
	Siltstone and chert, in alternating beds; siltstone is calcareous, gray, olive, maroon; weathers to shaly appearance; with abundant granular chert in even beds up to 6 in. thick, breaking into angular blocks upon weathering	90	
	Limestone; mostly covered	25	
			240
	Total thickness		1735

Table 2.15. Flows in Clinch, Emory, and Tennessee Rivers, 1945–1951[a]

Measured in cubic feet per second

	Clinch River						Emory River			Tennessee River					
	Miles 20.8 & 13.2			Mile 4.4[b]			Mile 12.8			Mile 529.9			Mile 465.3		
	Max.	Mean	Min.	Max.	Mean	Min.	Max.	Mean	Min.	Max.	Mean	Min.	Max.	Mean	Min.
January	22,900	8,960	1620	70,700	14,400	2120	50,000	4450	178	181,000	47,700	15,800	218,000	63,900	23,200
February	27,700	10,100	1230	88,800	15,800	2250	69,000	4550	468	204,000	50,800	17,900	195,000	67,900	19,700
March	12,700	5,850	690	26,700	9,450	1830	15,400	2910	507	100,000	32,400	13,300	148,000	44,200	19,500
April	8,540	3,400	306	13,300	5,620	752	6,600	1800	249	43,700	23,800	5,200	82,000	27,700	13,200
May	8,080	2,750	298	19,700	4,700	520	13,100	1610	58	38,300	20,000	3,000	95,900	28,800	17,900
June	7,420	2,820	224	9,280	3,320	262	5,300	396	14	32,100	18,900	8,200	32,800	25,500	18,900
July	7,630	2,930	259	12,800	3,400	281	9,230	360	21	87,500	19,600	6,500	106,000	26,400	15,400
August	8,390	4,520	374	8,760	4,800	378	3,060	177	4	37,600	21,400	9,900	43,300	28,100	17,800
September	8,450	4,620	341	13,000	4,940	462	5,500	224	2	39,900	22,200	6,900	54,400	30,100	17,100
October	9,200	5,130	150[c]	14,200	5,300	150[b]	5,040	93	1	67,100	23,500	9,800	72,800	29,700	16,300
November	12,700	4,430	453	40,500	5,830	556	27,800	1100	2	128,000	25,400	10,300	167,000	34,000	13,600
December	27,000	8,360	569	60,300	11,700	593	33,300	2720	24	112,000	41,000	11,800	139,000	53,600	21,100
October–April[d]		6,600			9,730			2520			34,900			45,900	
May–September[e]		3,530			4,230			553			20,400			27,800	

[a] *EGCR Hazards Summary Report*, ORO-586 (Oct. 10, 1962).
[b] Flows shown for Clinch River mile 4.4 include Emory River flows.
[c] By agreement with TVA, a flow of not less than 150 cfs has been maintained in the Clinch River at Oak Ridge since Aug. 28, 1943.
[d] Nonstratified flow period.
[e] Stratified flow period.

2.3.5 Seismology

Information on the frequency and severity of earthquakes in the East Tennessee area is reported in the *Aircraft Reactor Test Hazards Summary Report.*[9] Earthquake forces generally have not been considered important enough to be considered in the design of facilities either at ORNL or by the Tennessee Valley Authority (TVA) in this region. The Oak Ridge area is currently classified by the U.S. Coast and Geodetic Survey as subject to earthquakes of intensity 6, measured on the Modified Mercalli Intensity Scale.

Both Lynch[10] of the Fordham University Physics Department and Moneymaker[11] of TVA indicate that such earthquakes as occasionally occur in the East Tennessee area are quite common to the rest of the world and are not indicative of undue seismic activity.

An average of one or two earthquakes a year occurs in the Appalachian Valley from Chattanooga, Tennessee, to Virginia according to TVA records. The maximum intensity of any shock recorded is 6 on the Woods-Neumann Scale. A quake of this magnitude was experienced in the Oak Ridge area on September 7, 1956, and was barely noticeable by either ambulatory or stationary individuals. Structures were completely unaffected. Disturbances of this type are to be expected only once every few years in the Oak Ridge area.

The Fordham University records indicate a quake frequency below that of TVA. However, the magnitude of the observed quakes is approximately the same. Lynch indicates that "it is highly improbable that a major shock will be felt in the area (Tennessee) for several thousand years to come."

[9] W. B. Cottrell *et al.*, *Aircraft Reactor Test Hazards Summary Report*, ORNL-1835, pp. 78–79 (January 1955).

[10] Letter from J. Lynch to M. Mann, Nov. 3, 1948, quoted in a report on the *Safety Aspects of the Homogeneous Reactor Experiment*, ORNL-731 (Aug. 29, 1950).

[11] B. C. Moneymaker to W. B. Cottrell, private communication (Oct. 27, 1952).

Table 2.16. Community Water Systems in Tennessee Downstream from ORNL, Supplied by Intakes on the Clinch and Tennessee Rivers or Tributaries[a]

Community	Population	Intake Source Stream	Approximate Location	Remarks
ORGDP (K-25 area)	2,678[b]	Clinch River	CR mile 14	Industrial plant water system
Harriman	5,931[c]	Emory River	ER mile 12	Mouth of Emory River is at CR mile 4.4
Kingston Steam Plant (TVA)	500[d]	Clinch River	CR mile 4.4	
Kingston	2,000[d]	Tennessee River	TR mile 570	River used for supplementary supply
Watts Bar Dam (Resort village and TVA steam plant)	1,000[d]	Tennessee River	TR mile 530	
Dayton	3,500[c]	Richland Creek	RC mile 3	Opposite TR mile 505
Cleveland	16,196[c]	Hiwassee River	HR mile 15	Mouth of Hiwassee River is at TR mile 500
Soddy	2,000[d]	Tennessee River	TR mile 488	
Chattanooga	130,009[c]	Tennessee River	TR mile 465	Metropolitan area served by City Water Company
South Pittsburg	4,130[c]	Tennessee River	TR mile 435	
Total	168,061			

[a] *EGCR Hazards Summary Report*, ORO-586, Oct. 10, 1962.

[b] Based on May 1963 data.

[c] 1960 Report of U. S. Bureau of the Census.

[d] Based on published 1957 estimates.

3. BUILDINGS

3.1 Introduction

The ORR facility is composed primarily of three buildings: the reactor building, the reactor primary-water pumphouse, and the reactor secondary-water pumphouse and cooling tower. In addition to these, several small structures housing ancillary components are located throughout the area. They consist of structures such as the pool secondary pumphouse and cooling tower, the pressurizable off-gas (POG) filter pit, the cell-ventilation filter pit, the heat-exchanger pit, and the primary-coolant bypass-valve pit.

3.2 Reactor Building

The reactor building is a mill-type semiairtight steel-framed structure covered with insulated metal panels. The building covers an area 111 ft 8 in. long by 102 ft 10 in. wide. The floor level of the full basement is 20 ft below the first-floor level. The height of the 60-ft-wide center high-bay area is 64 ft 2 in. from the first-floor level to the bottom of the roof trusses. The second floor is actually at street level on the west end of the building and extends the full width and length of the first floor. It is 12 ft 6 in. above the first floor. The third floor is the top level in the low-bay areas, extending the full length of the building, 25 ft 6 in. above the first floor. It is not a solid construction covering the entire area of the building, but rather it is similar to a balcony running along the north and south sides and the west end of the reactor building. The building is windowless to eliminate glare on the pool surface and to aid in making the building more airtight under emergency conditions. Figure 3.1 shows a partial cutaway perspective view of the structure.

The interior finish of the building is similar to that used in light manufacturing construction.

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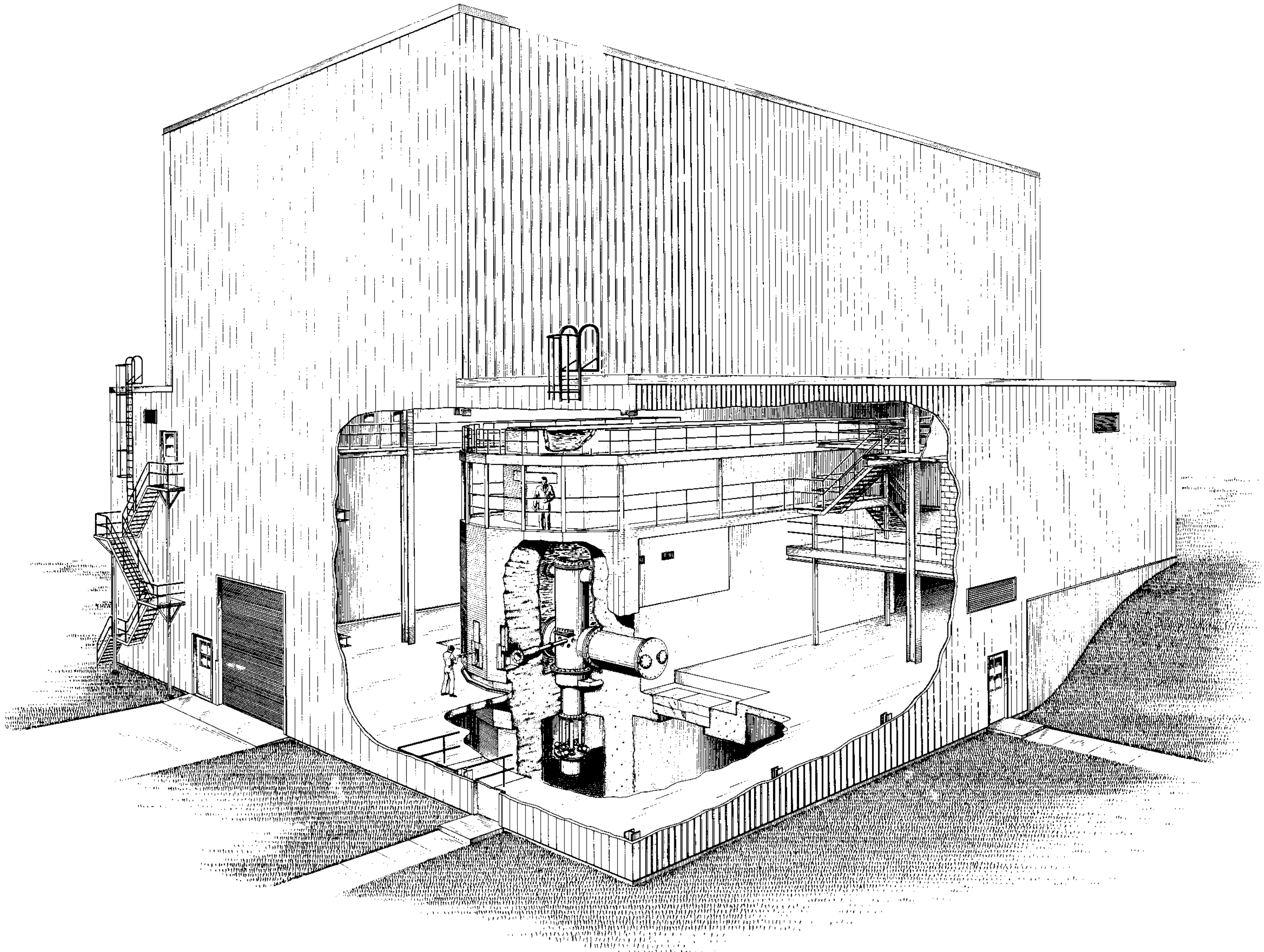


Fig. 3.1. Perspective and Cutaway View of ORR Building.

Roof beams and steel framing are exposed but painted. Utility conduit and pipe runs are exposed. Basement walls are painted only around the reactor substructure, and there only for better light reflection characteristics. Floor finishes are hardened concrete, integrally color coded for safe floor loading identification. The third-floor office and control-room wing has a refined construction finish only to the extent that the floors have asphalt tile. Air conditioning in this area ensures proper operation of the reactor control equipment. Toilet areas have quarry tile floors and painted masonry walls.

3.2.1 Basement

The basement floor space includes facilities for pool cooling, experiment cooling, and miscellaneous pumping operations, pool- and reactor-water demineralizer units, pool fill and drain pumps, electric-power distribution center, auxiliary ventilating fans and air filter banks, subpile room, plug storage facility, and experimental work and storage areas (see Fig. 3.2).

The water-system components are located underneath the storage pools to take advantage of the overhead shielding provided by the pool water. Shielding walls around the "hot" pumping equipment are of stacked barytes block with labyrinth doors where flexibility is necessary for removing or repairing pumps and of poured reinforced concrete where permanent walls are required. The basement sump (which is provided with a collection tank and a sump pump) is located in the water-system area. All basement drain lines lead into the sump collection tank. Since gravity drain from the basement to the plant process system is impractical, the sump pump discharge is directed into the first-floor drainage system.

The subpile room and adjacent pipe chase were designed on the basis of structural characteristics, which were required to support the reactor structure, and the wall thickness required to provide sufficient biological shielding. Due to the latter requirement, barytes concrete walls were used for the pipe chase to provide shielding from the radiation from the reactor exit cooling-water line. Since about 6 ft of barytes aggregate covering was required where the 24-in. cooling-water lines passed under the basement floor out from under the pools, the bottom of the pipe chase was located at an elevation 8 ft 6 in. below the basement floor. The subpile room footings could have been stepped up above the pipe chase footings to prevent excavating some rock; but, due to the sloping shelf rock formation prevailing in the area, this was deemed inadvisable. The 13 ft 6 in. by 10 ft subpile room has an 8 ft 6 in. basement which is necessary to allow removal of the control-rod drives from the reactor.

The subpile room houses the mounting plate for all the reactor control mechanisms. Certain utilities provided for use in experiments are described elsewhere. The pipe chase contains most of the cooling-system line interconnections and serves as a convenient location for cooling-system instrumentation. The lowest points of all cooling-system lines are in the pipe chase, and draining of all the lines for maintenance is performed at this point. The north wall of the pipe chase, through which the cooling-water lines pass, is constructed of stacked barytes concrete block above the first-floor level. A 4-ft-square manhole filled with stacked barytes concrete block is provided in the solid wall on the opposite side for pipe chase access. A removable instrumentation plug is located near the manhole.

Cylindrical cavities are provided in the basement wall for storing 32 plugs used in the reactor beam holes and for 6 somewhat larger plugs used in the large-facility beam holes during some stages of reactor and experiment operation. Each plug storage cavity has a connection to the normal off-gas suction line.

A water-collection tank is located in a pit at the west wall at the low point of the off-gas system. Condensate in the off-gas system drains into the tank and is discharged into the intermediate-level waste (ILW) line underneath the basement floor. The tank must be drained periodically by closing the valve in the line to the off-gas header, opening the line to the ILW system, and introducing compressed air reduced to 5 psig. A valve in the line from the off-gas header to the plug storage spaces is also located in this drain pit.

All basement construction is reinforced concrete except for the 12-in. concrete-block walls around the building-ventilation fan room and stairwells and the stacked-block concrete shielding walls around reactor components and experimental apparatus. The floor is designed for a live load of 1000 psf. Special service installations include a safety shower and two utility sinks.

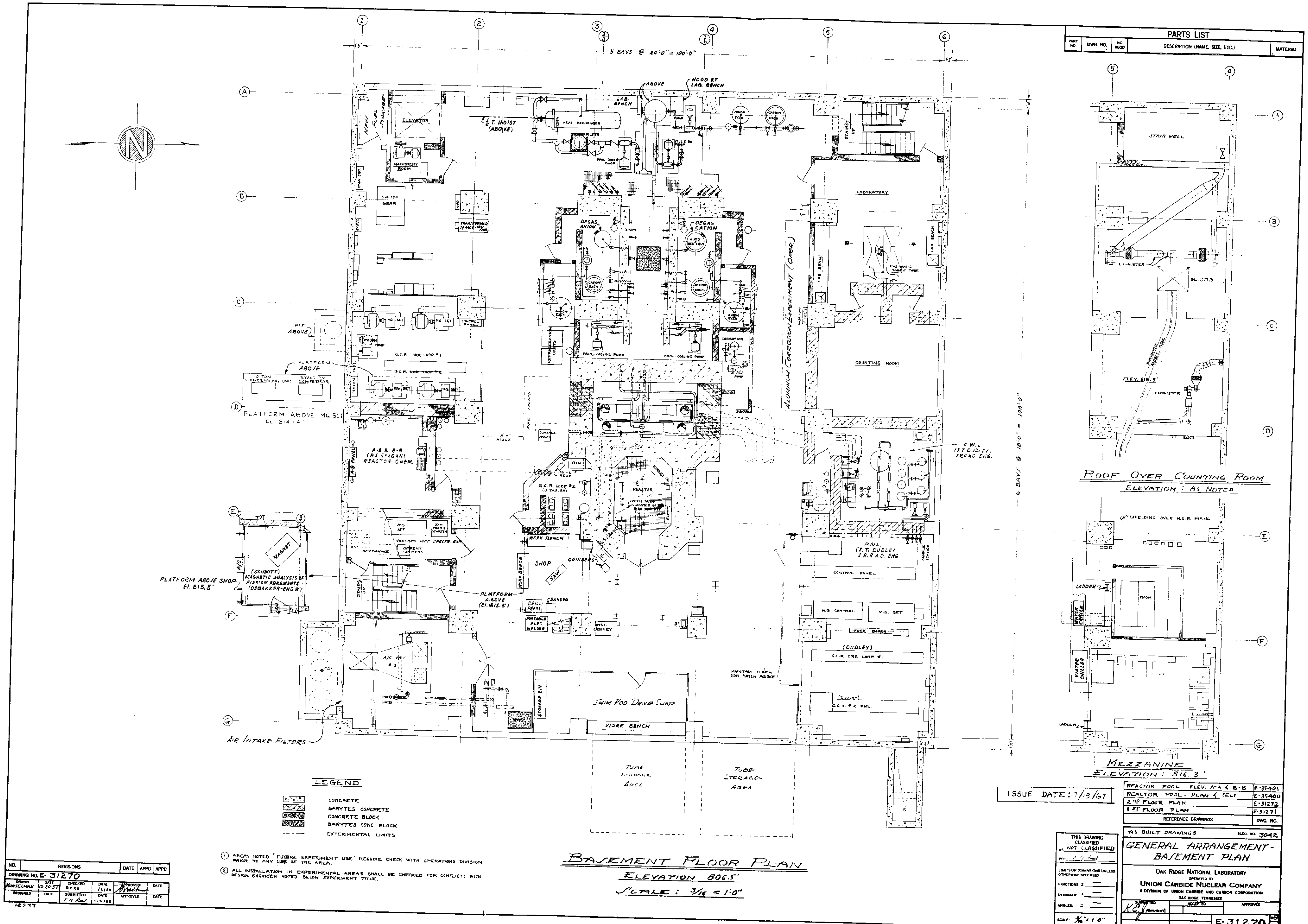


Fig. 3.2. Basement Floor Plan.

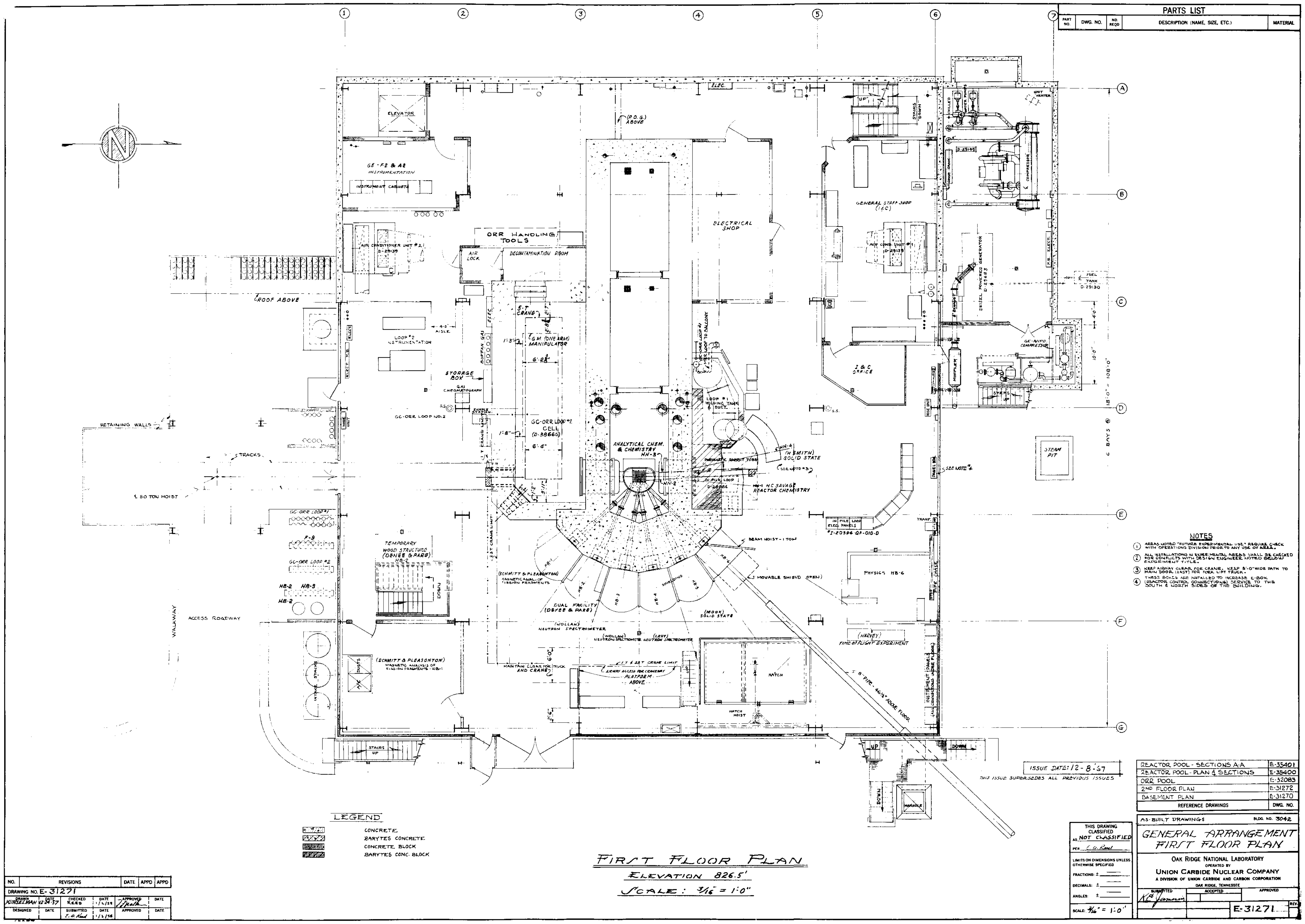


Fig. 3.3. First-Level Floor Plan.

3.2.2 First Floor

The entire floor area is free for research work except for the space occupied by the reactor structure, instrument shop, stairwells, elevator, and floor hatch for access to the basement. See Fig. 3.3 for a plan view of the first-floor area. The floor is of ordinary reinforced concrete construction and has a flat slab design except for framed openings. The slab thicknesses, live design load, and color codes are described in Table 3.1.

The reactor structure covers an area 18 ft wide by 55 ft long west of the reactor center line and a semicircle about 32 ft in diameter to the east. This structure is described elsewhere. Directly in front of each of the two large-facility openings in the reactor structure is an 8- by 10-ft hatch to the basement in a depression in the floor slab 17 ft 2 in. long, 11 ft wide, and 2 ft deep. The purpose of this design was to allow a cell of stacked-block construction to be built around an experimental rig at the large-facility location on the first floor and to make it possible to lower this rig by remote control into a second shielded cell in the basement for dismantling or for storage during radiation decay. Concrete plugs fitting flush with the depressed slab surface are provided for the hatch openings.

Numerous sleeves are provided through the first floor for connecting instrumentation, experimental apparatus, and utilities connections with the basement. Four-inch-diameter sleeves on about 9-ft centers are located around all the walls. Six 12-in. sleeves are located about 15 ft from the face of the reactor structure, 2 ft off each beam-hole center line. Four additional 12-in. sleeves are located near column line 2 in the beam-hole experimental area for general purpose use. Four 6-in. sleeves are located in the depressed floor slab at each large-facility installation and two additional ones convenient to the area around the floor depressions. Seven 6-in. sleeves and eight 12-in. sleeves are spaced at the base of the storage pools.

The 4-in. sleeves along the walls around the building project 3 in. above the floor level. All other sleeves are fitted with sealed covers that fit flush with the floor.

A 10 ft by 16 ft 6 in. hatch with a hinged steel cover provides access to the basement with the bridge crane. This hatch cover normally remains closed for fire-retarding purposes. There is a

Table 3.1. Floor Loading Specifications – First Floor

Area[a]	Thickness (in.)	Design Load	Color Code
2 bays A-B × 1-3	7	200 psf uniform or a 5-ton concentrated load on 6 ft^2 in any bay	Terra cotta
$\frac{1}{2}$ bay A-B × 3-4	7		
4 bays A-C × 4-6	7		
2 bays B-D × 1-2	7		
1 bay C-D × 5-6	7		
3 bays B-E × 2-3	54	Floor designed for 3500 psf uniform load; beams designed for $\frac{1}{3}$ this value	Brown
1 bay D-C × 1-2	11		
2 bays C-E × 4-5	54		
1 bay D-E × 5-6	11		
3 bays E-F × 2-5	11	10-ton concentrated load on 10 ft^2 adjacent to beam holes plus a 20-ton rolling load	Red
5 bays F-G × 1-6	11	20-ton[b] rolling load	Gray
1 bay E-F × 1-2	11		
1 bay E-F × 5-1	11		

[a]See Fig. 3.3.

[b]The 20-ton design rolling load is based on a 5-ft-long row of steel wheels carrying 16 tons and 4 tons concentrated on one steel wheel 7 ft from the main axle. This is an approximate description of the beam-hole-plug shield.

truck door in the east wall, 10 ft wide by 16 ft high, to allow entry of equipment that is to be used in the beam-hole experimental area.

A 4-in. curb all around the first-floor area is intended to prevent area contamination in case of a leak in the reactor structure or a water release from any other cause. There is a 4-in. ramp at all first-floor doors. This curb will direct water flow to the basement, except in the case of a catastrophic pool leakage.

Personnel doors are located in each side of the east wall of the building; the personnel doors that are approximately in the middle of the north and south walls are intended as emergency escapes and are not used for normal pedestrian traffic.

Special service installations include two service sinks with cold water only, two safety showers, and a drinking fountain.

3.2.3 Second Floor

The second-floor level coincides with the ground level on the west side of the building. The floor space is occupied by offices, men's and women's toilet facilities, a janitor's closet, and a truck-loading area to accommodate equipment to be placed in the storage pools. See Fig. 3.4 for a floor plan. The floor is a reinforced ordinary concrete slab on steel framing with the slab thicknesses, live design load, and color coding shown in Table 3.2.

The janitor's service closet contains a service sink with hot and cold water, a hot water heater for the toilets, and space for cleaning equipment, floor polishers, etc.

Four-inch sleeves through the floor for service connections to the first floor have been provided along the west wall, similar to those on the first floor.

Walls around the toilet area are designed to be permanent and are of 8-in. concrete-block construction. Walls around the second-floor offices are of the removable metal partition type, as some degree of flexibility of area enclosure is desirable in this region.

Personnel entrance doors are provided in each side of the west wall. Steps up to the balcony walkway around the reactor structure are located on each side of the pool. The only special service installation on the second floor is one safety shower.

3.2.4 Third Floor

On the third floor are the reactor control room and four offices on one side of the high-bay area, laboratory working space on the other side of the high-bay area, a hot cell which is constructed at the west end of the reactor structure, and a walkway all around the reactor structure (see Fig. 3.5). A steel walkway extends from the control-room wing of the third floor to the third-floor-level walkway around the reactor structure. Adjacent to this walkway are steel stairs down to the balcony walkway around the reactor structure. Although these walkways make for awkward maneuvering of the pendant-controlled crane, this inconvenience is not serious and does not warrant providing removable stair and walkway connections.

Table 3.2. Floor Loading Specifications – Second Floor

Area[a]	Thickness (in.)	Design Load	Color Code
1 bay A-B × 1-2	4	75 psf	Tan
2 bays A-B × 2-4	$7\frac{1}{2}$	200 psf or a 5-ton concentrated load on 6 ft^2	Terra cotta
2 bays A-C × 4-5	$8\frac{1}{2}$	25-ton highway truck or a 10-ton concentrated load on 10 ft^2	Maroon
2 bays A-C × 5-6	$4\frac{1}{2}$	75 psf	Tan[b]

[a]See Fig. 3.4.

[b]The toilet areas have a quarry tile floor.

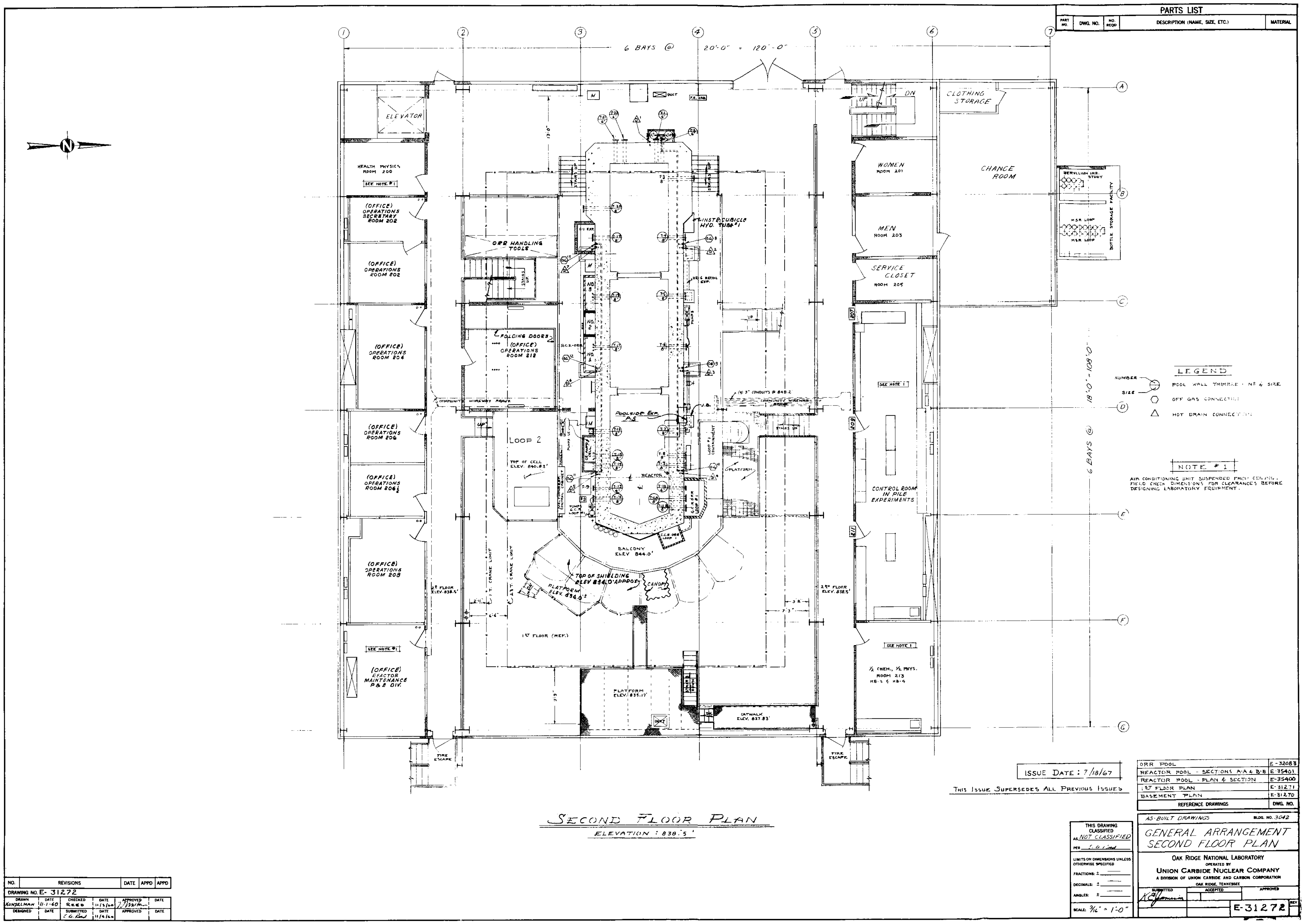


Fig. 3.4. Second-Level Floor Plan.

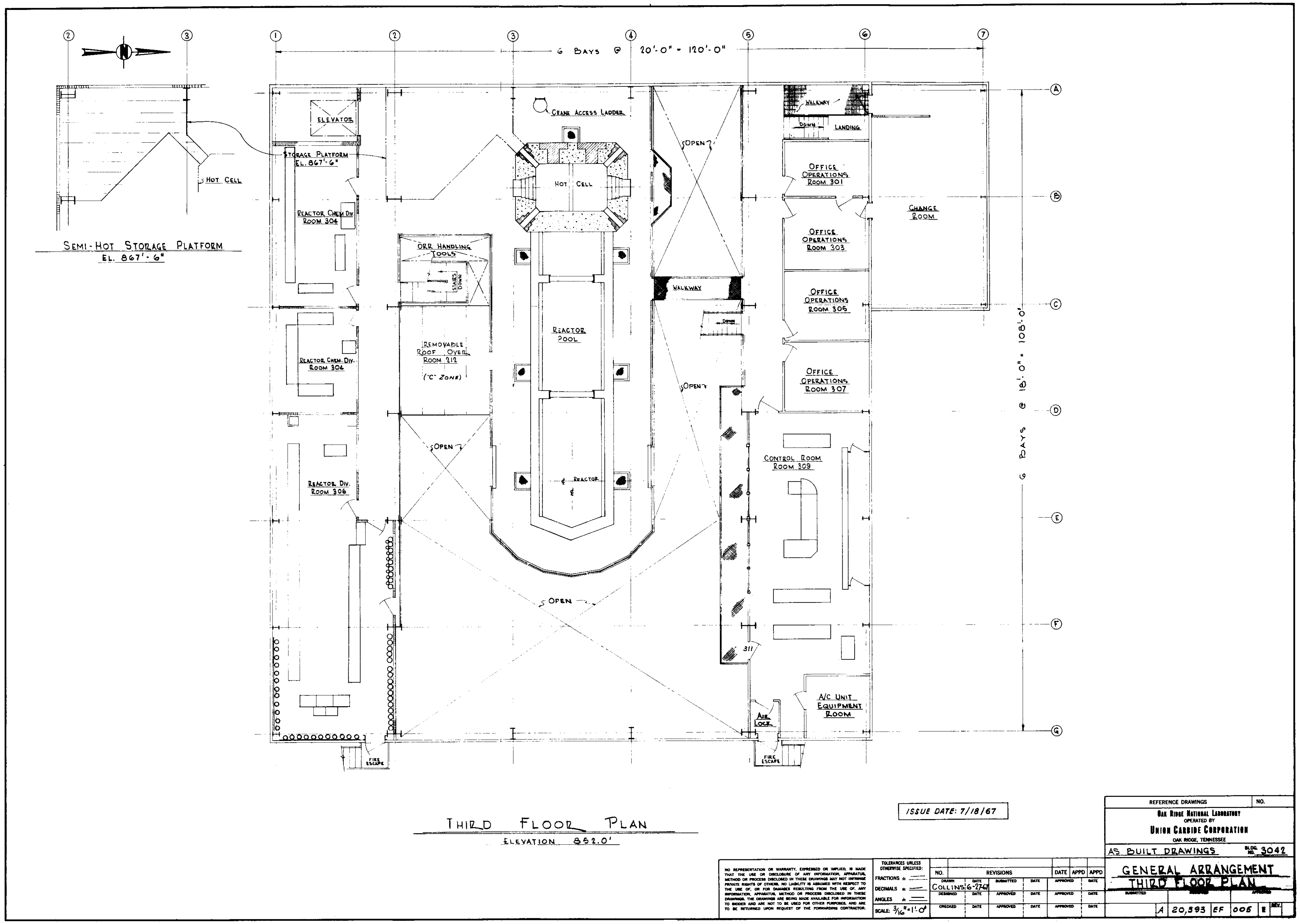


Fig. 3.5. Third-Level Floor Plan.

Table 3.3. Floor Loading Specifications – Third Floor

Area[a]	Thickness (in.)	Design Load	Color Code
6 bays A-G × 1-2	7	150 psf uniform load or a 3-ton concentrated load on 6 ft^2 in any bay	Tile red
3 bays A-D × 5-6	4	75 psf uniform load	No color code[b]
3 bays D-G × 5-6	4	150 psf uniform load	No color code[b]
1 bay A-B × 2-3	$7\frac{1}{2}$	160-ton uniform load or 5-ton concentrated load on 10 ft^2 in any bay	Terra cotta

[a]See Fig. 3.5.
[b]These bays have an asphalt tile floor covering.

Furthermore, the location and size of the reactor structure itself make special treatment of the crane control pendant necessary.

All of the third floor is of reinforced ordinary concrete construction on steel framing, with slab thicknesses, live design load, and color coding as shown in Table 3.3.

Since no concept of the specific nature of the work to be done in the laboratories in the building was available at the time of its design, no conventional laboratory equipment or furniture was initially installed. However, building utilities headers are provided in the rooms and have tees in the lines where connections may be made to them.

The laboratory area is divided into individual rooms by removable-type insulated metal partitions, which can be rearranged to suit work-space requirements, if necessary. This same type of removable panel is used to partition the four offices and the control room. Insulated panels were chosen because the control room, the offices, and the laboratories are air conditioned. The front of the control room contains plate-glass sections in the wall to enable the reactor operator to have a full view of the reactor structure and to provide a visitors' observation area. The glass panels have a 5° slope to eliminate glare from the building lights.

The area at the west end of the reactor structure, where the hot cell is located, is designed to support 3-ft-thick barytes concrete shielding walls around the cell. The inside dimensions of the cells are 3 ft 6 in. wide by 7 ft long by 7 ft high.

Four-inch pipe sleeves similar to those on the first floor are located in the floor along the outside wall of the laboratory wing and the hot cell area. Floor slots of varying widths with steel cover plates are provided through the floor along the back wall of the control room and next to the adjacent offices. In addition to the conduit sleeves in the control-room floor required for initial installations, additional sleeves have been provided through the control-room floor where possible future control expansion is envisioned. These extra sleeves have plugs flush with the concrete floor and are covered by the asphalt tile.

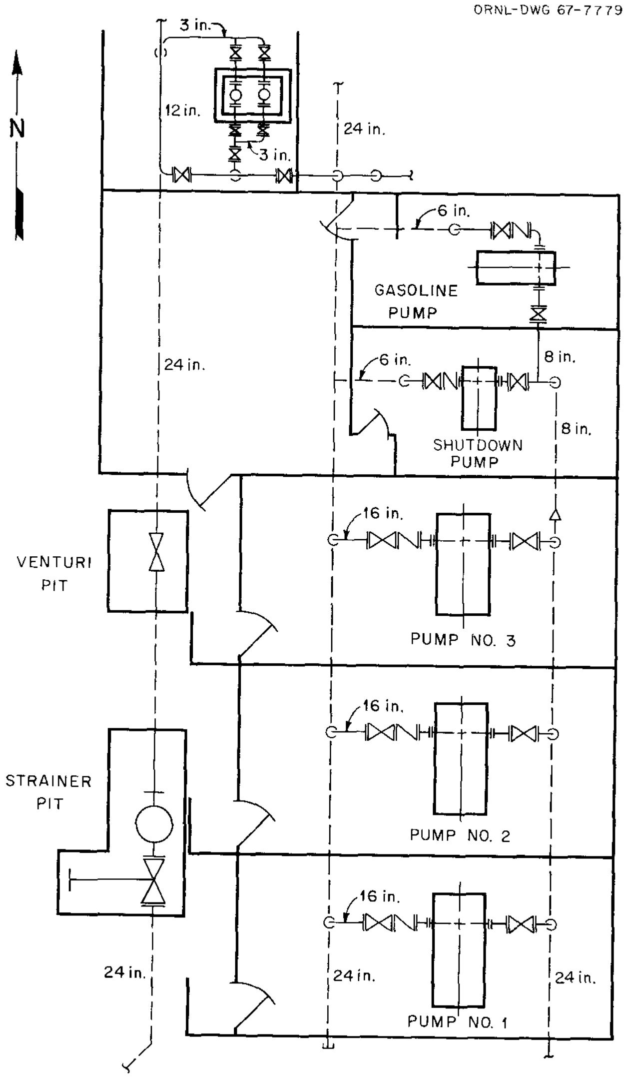
Special service installations include an emergency shower in the hot-cell area, two drinking fountains, and a service sink. Outside stairs for emergency exit are located at the east end of both the laboratory and control-room wings. The crane access ladder terminates on the third floor.

3.3 Auxiliary Buildings

3.3.1 Primary-Coolant Pumphouse

The primary-coolant pumphouse is a concrete-block structure located approximately 300 ft northeast of the reactor building. The overall size of the building is 68 ft by 42 ft. It is divided into five pump cells and an electrical

Fig. 3.6. Schematic of Primary Pumphouse Floor Plan.



equipment room. The three main pump cells are 30 ft 8 in. by 14 ft 8 in. The shutdown pump cell is 11 ft 8 in. by 22 ft 8 in., the cell for the emergency gasoline-operated pump is 11 ft by 22 ft 8 in., and the electrical equipment room is 23 ft 8 in. by 19 ft 4 in. A floor plan of the building is shown in Fig. 3.6.

3.3.2 Cooling Towers

Reactor Cooling Tower. – The reactor secondary coolant is cooled by a Marley model No. 669-3-02 induced-draft single-flow cooling tower. Figure 3.7 shows an end view of the tower. Just adjacent to the tower is the secondary-system pumphouse. Figure 3.8 shows a plan view of the building.

Pool Cooling Tower. – The pool secondary coolant is cooled by a United model No. 1818-AM16-10-15 induced-draft double-flow cooling tower. Adjacent to the tower is the secondary-system pumphouse, which houses the electrical switchgear and chemical treatment equipment.

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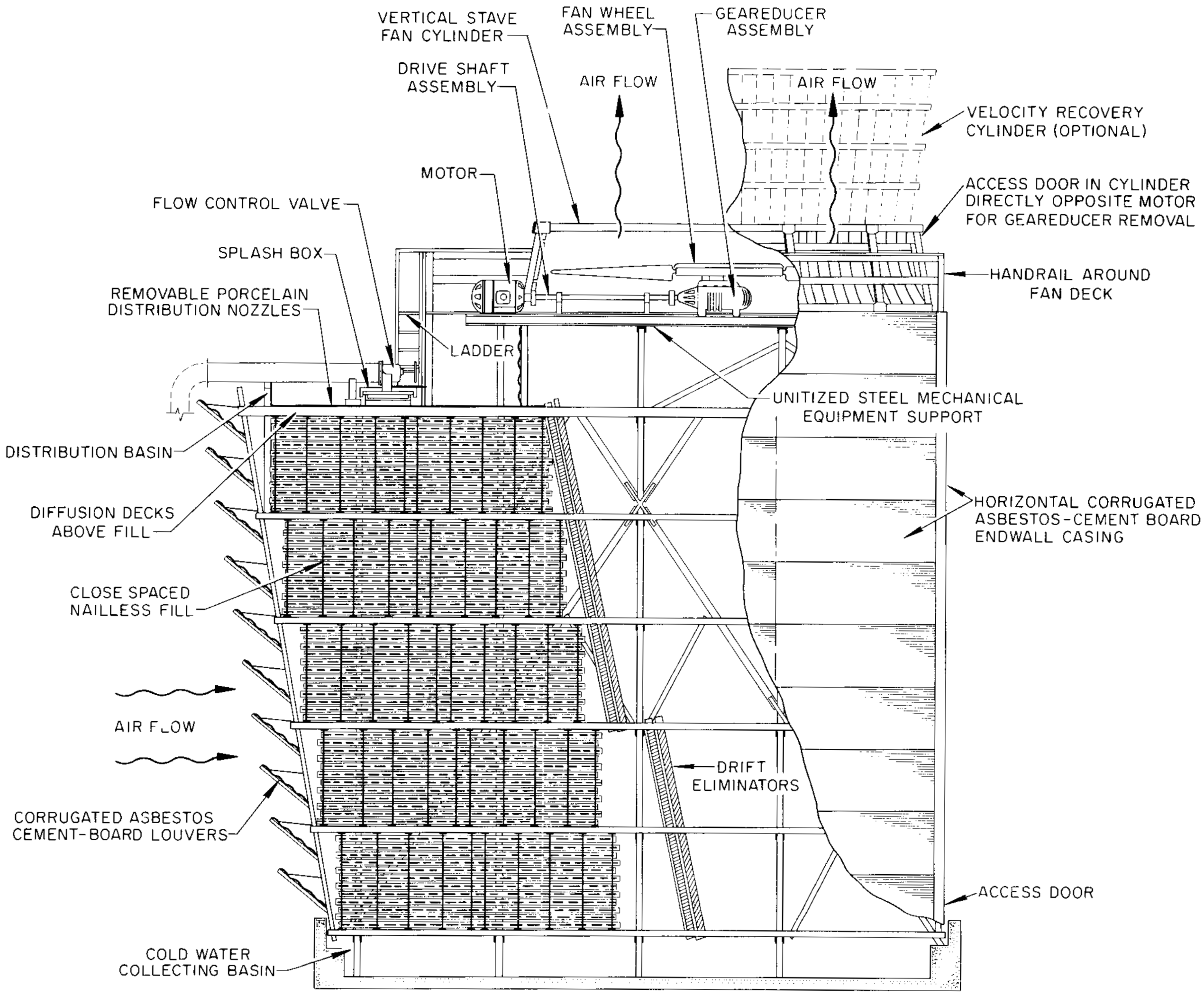

Fig. 3.7. End View of Secondary Cooling Tower.

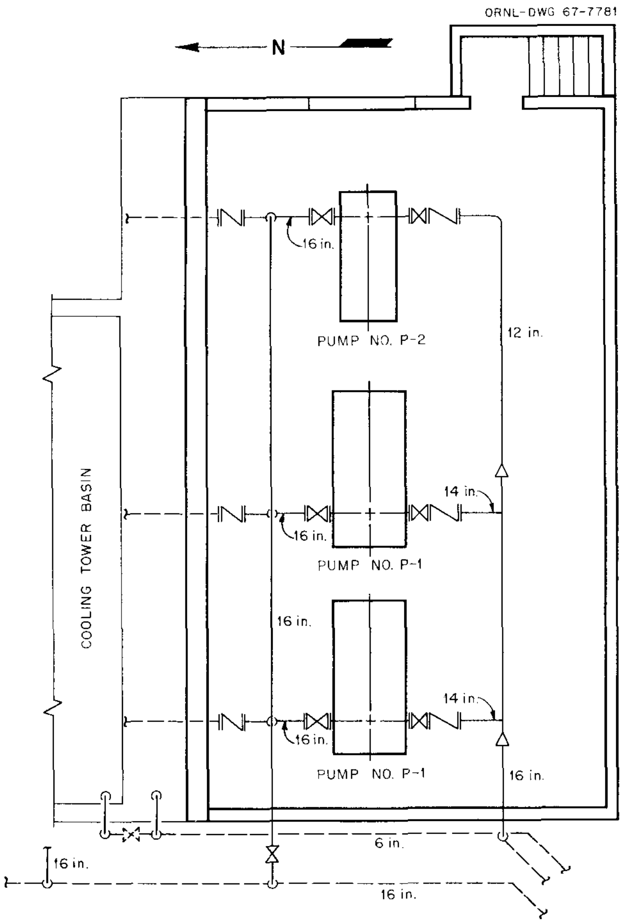

Fig. 3.8. Schematic View of Secondary Pump House.

4. CONTAINMENT, VENTILATION, AND AIR CONDITIONING

4.1 Introduction

The ORR containment, ventilation, and air-conditioning systems have been designed to minimize the dissemination of contamination and radioactive gases into the area surrounding the reactor building in the event of an accidental release of radioactive materials. There are three separate systems which are used for the decontamination and disposal of potentially radioactive gases. The cell-ventilation system provides dynamic containment for the building and provides a method for the decontamination and subsequent controlled release of large quantities of potentially radioactive gases. Two separate off-gas systems are used to handle routine low-volume disposal of gaseous wastes from experimental facilities and reactor system components. The two systems are the pressurizable off-gas system (POG) and the normal off-gas system (NOG). Although the off-gas systems are not specifically a part of the building containment system, they are somewhat related and are described in this section to aid in the overall understanding of the air-handling systems. The control room, certain offices, and other normally clean areas are served by an air-conditioning system which is separate from the building's ventilation system.

Additional ventilation of the reactor bay and experimental areas is provided by two roof fans, each of which exhausts building air at the rate of 7000 cfm. These automatically controlled fans also aid in maintaining the negative pressure in the building during normal conditions.

Ventilation of the basement area is provided by a centrifugal blower, which exhausts basement air through a louvered opening located in the northeast corner of the basement.

The cell-ventilation system provides a flow of air through areas of the main ORR building in which potential sources of radioactive gases exist. These areas include the reactor-bay area, the basement area, the Loop 2 decontamination cell and equipment room, the pipe chase, the hot cells over the reactor pool, the B-9 experiment cubicle in the basement, and other areas. The air from these areas is drawn through ducts into a filtering system and discharged into the atmosphere from the 250-ft-high 3039 stack.

The off-gas systems are designed to handle routine high-concentration radioactive gaseous releases and are connected directly to the components which are capable of releasing these effluents. The pressurizable off-gas system is connected to components which may become pressurized, whereas the normal off-gas system collects gaseous wastes only from components which are not subject to becoming pressurized.

4.2 Containment

The ORR employs the concept of dynamic containment to prevent the escape of radioactive gases to the atmosphere.[1] The ORR building is not sealed airtight to retain radioactive gases that might be released. Instead, it is maintained as a partially leak-tight structure. The cell-ventilation system maintains the containment by the controlled exhaust of air from the building at a rate sufficient to ensure that there is always an inflow of air at leakage points. The cell-ventilation system is not a start-on-demand system, but rather it operates continuously, so that the building is always under a slight vacuum.

When the building ventilation is placed in the emergency containment mode, all air entering the building enters by leakage paths. Heating and ventilating units, exhaust fans, and air-cooling equipment are shut down, and their related louvers are automatically closed. At the same time, the roof fans and the basement exhaust fans are shut off, and their louvers are closed so that all air leaves the building through the ducts of the cell-ventilation system. The air exhausted from the building is decontaminated by a filtration system and discharged into the atmosphere at a height sufficient to ensure atmospheric dispersion.

The building ventilation can be placed in the containment mode either manually or automatically through the use of three separate sets of controls. Within the first set of controls, designated as containment controls, manual actuation is initiated by a push button located in the control room. Automatic containment will occur when a high radiation level exists either in the building or in the cell-ventilation duct, as detected by the containment radiation monitors. A detector located just outside of the control room initiates containment if a level of 75 mr/hr exists. A detector on the cell-ventilation duct at the filter pit also initiates containment if a level of 7.5 mr/hr is detected. The two radiation detectors transmit signals to two electrometers located in the control room, and containment is initiated when set points on the electrometers are exceeded. The ventilation duct electrometer is a fail-safe unit in that containment is also initiated when the instrument fails.

Building containment can also be effected by the facility radiation and contamination alarm system (Sect. 8.7). This system will automatically actuate the building evacuation system if two monitrons or two continuous air monitors in the coincidence circuit alarm simultaneously at the high-level trip (23 mr/hr and 4000 counts/min respectively). Also, containment can be actuated by any one of three "manual" evacuation buttons (two in the control room and one outside a west personnel door).

The third method of automatically establishing building containment is provided by the fire-alarm system. If a fire alarm occurs in the building, containment is initiated by a relay in the building's master fire-alarm box.

In order to restore normal operating conditions, all of the building heating and ventilating controls must be manually reset. The truck doors can be opened with the building under containment and without the reset action; however, when this is done, an operator stands by the doors to close them if an emergency demands such action.

To assure that the personnel in the ORR control room and offices on the third level, north, have an independent air supply when the building is in containment, the air-conditioning unit serving these rooms does not exchange air from the building high-bay area with the office and control-room area. Instead, the unit exchanges air from the outside. The rooms are maintained at a pressure greater than that existing in the high-bay area.

Pressure conditions within the building are continuously indicated by four manometers located in various parts of the building and by flow indicators which monitor the exhaust flow in the cell-ventilation duct. A reduction of the duct flow below 2500 cfm results in a reactor power reduction to 300 kw.

4.3 General Ventilation and Air-Conditioning Systems

Ventilation and air conditioning of the ORR building are accomplished with several package-type air conditioners positioned in appropriate locations in the building. Chilled water is supplied to the various air-conditioning units by a single large unit which has a capacity of 2.15×10^6 Btu/hr (180 tons). Heat collected at the chiller is dissipated by a small cooling tower, which is used for that system only. Low-pressure steam (25 psi) for the air-conditioning units is obtained

[1]F. T. Binford and T. H. J. Burnett, *A Method for the Disposal of Volatile Fission Products from an Accident in the ORR*, ORNL-2086 (Aug. 2, 1956).

by a pressure reduction of the 125-psi plant steam supply.

The basement is heated by a forced-draft unit with steam-fed heating coils and a design specification capacity of 12,000 cfm. This unit has pneumatically operated inlet and exit dampers and is located near the basement ceiling on the south wall. Fresh air for the unit enters through a metal stack located outside on the south side of the building. Air from the basement is exhausted through a vertical shaft located in the northeast corner approximately 10 ft above the basement floor.

The subpile room, located at the east end of the basement under the reactor pool, is ventilated and air conditioned by a window-type unit mounted in the east wall of that room. Air from the subpile room is exchanged with that from the basement.

The heating and ventilation system of the laboratory and the counting room located in the northwest corner of the basement consists of air-conditioning facilities in both rooms. The units supplying these rooms can supply climatized air. The unit servicing the counting room is backed up by a booster fan, served by a duct from the fresh-air intake to the No. 2 heating and ventilation unit, to maintain this room at a positive pressure with respect to the rest of the building.

The laboratory air is exhausted both by hood fans, serving two laboratory hoods, via an exhaust duct which passes directly through the building roof to the atmosphere and by natural air movement to the building basement through a hole in the laboratory ceiling. The laboratory hood exhaust fans, which are vented to the atmosphere, are controlled by switches located on the lower sections of the hoods and must be reactivated after a building ventilation shutdown.

The heating, cooling, and ventilation of the rooms and open area of the first floor are accomplished by the Nos. 1 and 2 air-conditioning and ventilating units located on this floor. Each of these has a capacity of 15,600 cfm. The No. 1 unit is located near the south wall of the building, and the No. 2 unit is located near the north wall of the building. These units are supplied with chilled water and steam to the cooling and heating coils respectively. They are thermostatically controlled with automatically controlled inlet and recirculation air dampers. Their fresh-air inlets are located above and adjacent to the south and north personnel doors. Each unit has ducting, equipped with diffusers, which runs east and west from the unit on its respective side of the building. Each unit is equipped with filters for dust removal.

The heating, cooling, and ventilation of the second-floor open area and the open area of the rest of the building not served by the first-floor units are accomplished by air-conditioning and ventilating unit No. 3. This unit has a design specification capacity of 61,200 cfm. As with units Nos. 1 and 2, chilled water and steam are supplied to its coils. It has pneumatically operated inlet dampers, and the fresh-air intake is by way of three metal stacks located outside the building at the southeast corner. The intake air is filtered for dust removal.

The laboratories and offices of the second level, north and south, contain individual heating and cooling units equipped with coils supplied by chilled water and steam. These units are thermostatically controlled and have dampers controlling the amount of recirculated air from the room and fresh makeup air from the open area of the building outside the room. An exception to this type of cooling is Room 212, which is equipped only with an exhaust fan, exhausting to the open area of the second floor.

The second-floor change room is heated by a forced-draft unit. This unit has thermostatically controlled dampers. It uses either recirculated air from the room or fresh air obtained from the outside through an intake on the west wall of the building at the unit. Air is removed from the room to the outside of the building by two thermostatically controlled exhaust fans in the east wall.

The second-floor rest rooms are vented through a duct, with air-intake registers of 400 cfm each, to an 800-cfm fan which exhausts through a self-closing louvered opening in the east wall of the change room.

The control room and offices of the north side of the third level are heated, cooled, and ventilated by an air-conditioning unit located in the northeast corner of the third floor. This unit has a design specification capacity of 4460 cfm, is equipped with its own refrigeration system as well as heating system, is thermostatically controlled, and has pneumatically controlled automatic inlet and return-air dampers. Its fresh-air intake is located in the north wall of the building at the unit.

The south section of the third floor consists of laboratories and offices, which contain individual heating and cooling units. Each unit, equipped with coils supplied with chilled water and steam, is thermostatically controlled and has automatically operated dampers for either recirculation of room air or makeup with fresh air from the open area of the third level, similar to those on the second level. Rooms 306 and 308 now have window-type air conditioners exhausting to the open area of the third floor. In addition, one of the previously described units is located in the open area outside of Room 310 and is ducted with appropriate diffusers to Room 310, which is now a single unit with Rooms 306 and 308. The laboratories have individual roof vents, the openings of which are pneumatically operated and controlled.

The third-floor change room is heated and ventilated by a thermostatically controlled system which is similar to that of the second-floor change room.

Air from the first, second, and third floors and the open area, which extends to the building roof, is exhausted by two of six roof exhaust fans in addition to the building ventilation system previously described. Two of these fans, Nos. 1 and 6, have capacities of 7000 cfm each and can be operated in an automatic mode. The remaining four exhaust fans have capacities of 16,000 cfm each and are manually operated. At present, only exhaust fan No. 3 is occasionally called upon to assist fans 1 and 6 to maintain the required building negative pressure while the normal building air intakes are open. Fans 1 and 6 are currently operated in the automatic mode in conjunction with the building ventilation system. All of these fans are equipped with pneumatically operated, automatically controlled dampers.

4.4 Cell-Ventilation System

The cell-ventilation system is a maximum-reliability system which provides for the removal, decontamination, and disposal of possibly contaminated air from the reactor building (Fig. 4.1). It consists of a network of five ducts located throughout the building, a series of filters which remove particles and radioiodine from the air, blowers which provide the draft for the system, and the 250-ft-high 3039 stack. This system removes air from the building at a rate of 9000 cfm.

The five main ducts and the areas they service are listed below.

1. The main building duct removes air from:
   a) around the pool area between the second and third floors,
   b) the overhead reactor-bay area,
   c) the basement area over the B-9 cubicle.
2. A basement cell duct removes air from the Loop 2 decontamination cell and equipment room, which is located on the first floor.
3. A basement cell duct removes air from:
   a) the NOG seal tank in the subpile room,
   b) the HB-5 vacuum pump,
   c) the pressurized-water loop (PWL) battery room,
   d) the PWL valve box,
   e) the PWL equipment room,
   f) the pipe chase,
   g) the pipe-tunnel sump and sampling stations.
4. The hot-cell duct removes air from the hot cells over the reactor pool.
5. A duct in B-9 cubicle removes air from the B-9 cubicle in the basement.

Each duct has a manual damper, which is adjusted to provide suitable air flow and pressures and is then locked at the required settings. All five ducts join one 24-in. duct before entering the filter pit. The filter pit (Fig. 4.2), located south of the reactor building, is an underground waterproofed concrete structure designed to withstand pressures in the range -57 in. H_2O to $+5$ psig. The top shield of the filter pit is $1\frac{1}{2}$ ft thick, and removable concrete plugs allow access to the filters.

The filter pit contains five separate banks of filters. They are, in order, a 3×3 array of 2-ft-square Fiberglas prefilters, a 3×3 array of 2-ft-square Fiberglas absolute filters, two 3×3 arrays of 2-ft-square $\frac{3}{4}$-in.-thick activated charcoal filters, and another 3×3 array of 2-ft-square absolute filters. The filter frames are of corrosion-protected carbon steel.

The filter pit is designed for a filter-removal method in which the filters are withdrawn into lead shielding casks. After removal of the concrete access plugs, the cask is placed over the opening. Wedging devices which seal the filters to the filter frames are removed remotely, and the filters are removed into the casks.

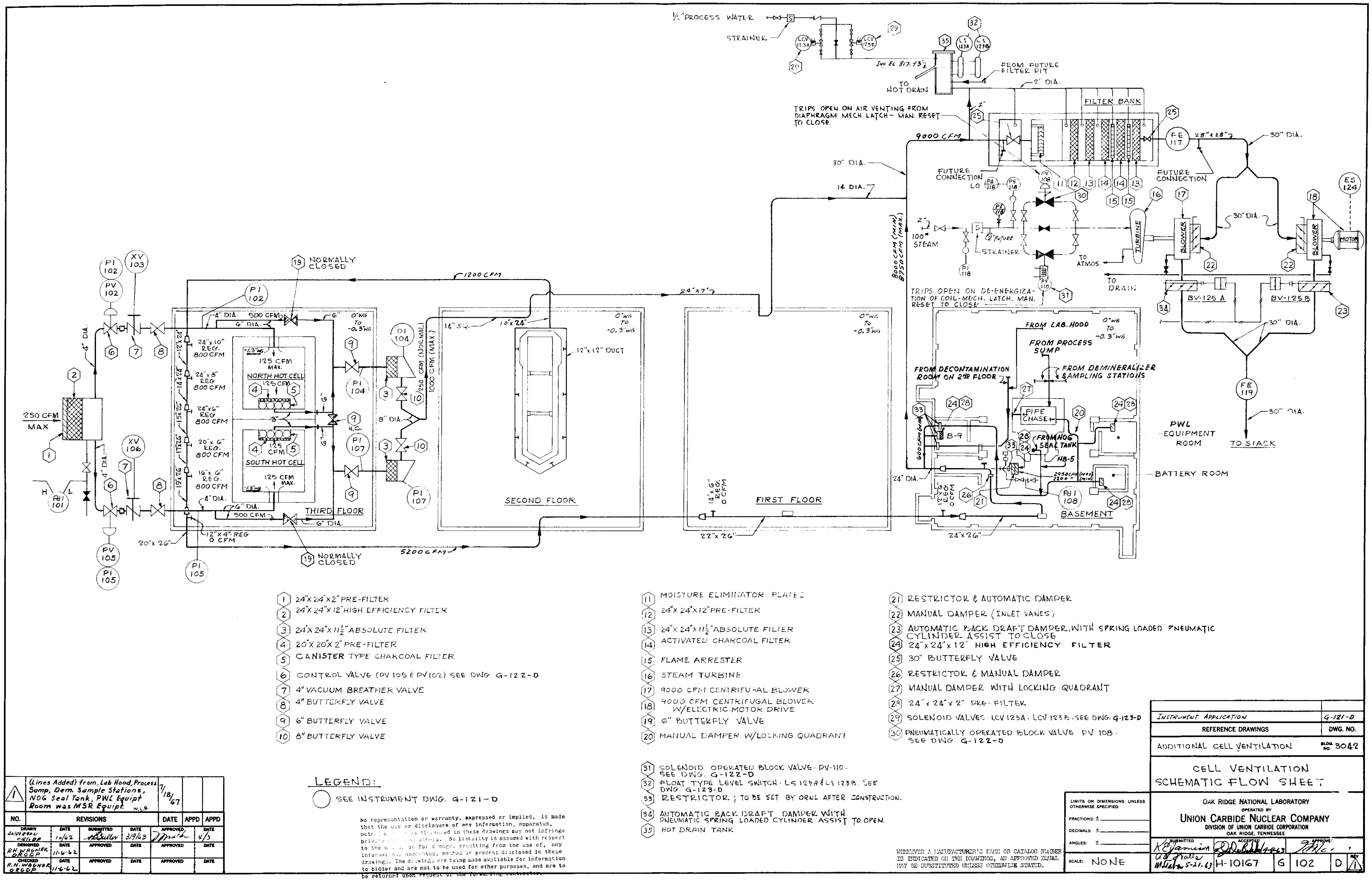

Fig. 4.1. Schematic Diagram of Cell-Ventilation System.

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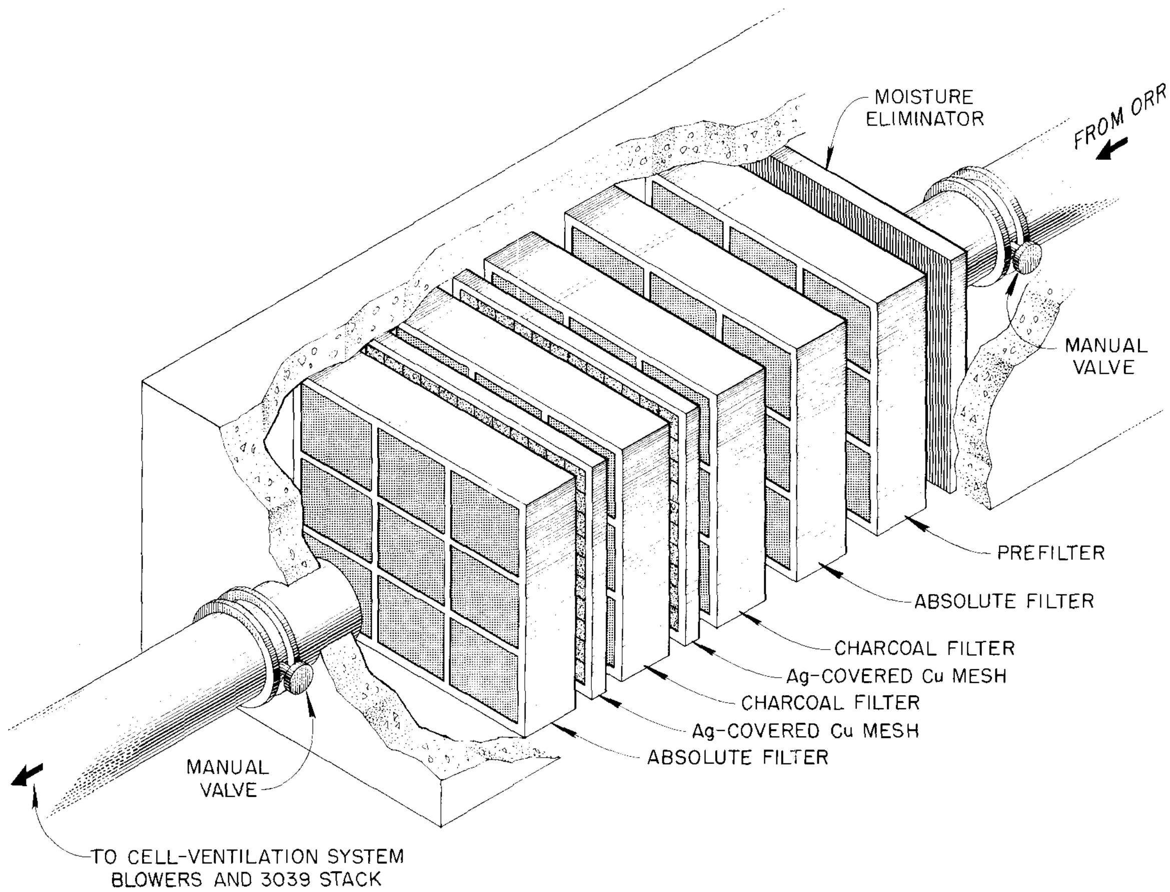


Fig. 4.2. Cell-Ventilation Filter Pit.

Pressure differentials across each bank of filters are indicated by gages located at the filter-pit area. An additional gage, indicating the flow rate through the filters, is also provided.

Since charcoal has a relatively low combustion point and would be subject to heat should fission products or other radioactive substances be deposited on it, the possibility of a fire in the charcoal filters exists. Accordingly, silver-covered copper-mesh flame arresters were installed downstream of each bank of charcoal filters, and thermocouples were installed to warn of increasing temperatures (alarm at 38°C). These temperatures are continuously recorded by multipoint recorders in the control room.

Each filter bank has an intermediate-level waste drain line to prevent any accumulation of water. These drain lines are connected to an underground seal tank, and a water leg is maintained in this tank to provide an airtight seal between the filter system and the drain. The water also seals the individual compartments from each other.

Manually operated butterfly-type shutoff valves are located on the inlet and exit sides of the filter pit. When neither of the cell-ventilation blowers is in service, these two valves are the only valves capable of absolutely preventing backflow from the stack into the ORR building.

Downstream from the filter system are located the two blowers (in parallel) which maintain the building's negative pressure at about -0.3 in. H_2O by removing air from the building at a rate of 9000 cfm. Normally, an electrically powered blower of 40,000 cfm capacity is connected in such a way as to provide the 9000-cfm draft for the ORR building. The remaining capacity is used elsewhere in the Laboratory. An emergency steam-driven blower is maintained as an emergency standby unit and provides the necessary draft for the building in the event of a failure of the electrically powered blower.

4.5 Off-Gas Systems

4.5.1 Normal Off-Gas System (NOG)

The NOG system (Fig. 4.3), with some modifications, has been in operation since the initial operation of the reactor. It consists basically of a piping system embedded in the concrete of the pool wall and building floor with numerous access ports, two water catch tanks to prevent accumulated liquids from blocking the flow channels, and an 8-in.-diam stainless steel line from the ORR building to the 3039 stack area, where the gases pass through a scrubber cleaning system before being discharged through the stack. Only ORR components that are incapable of pressurizing the NOG system are connected to this piping system.

Under normal conditions, the NOG is kept at a negative pressure of ~ -40 in. H_2O by an electric-motor-driven positive-displacement blower. In the event of an electrical power failure, or if for any other reason the pressure in the intake manifold increases to -30 in. H_2O, an auxiliary steam-turbine-powered blower will be energized automatically to provide continuation of the off-gas service.

The "cleanup system" for the NOG is a continuously operating recirculating scrubber using 1% NaOH and a filter bank consisting of a roughing filter, an absolute filter, and a metallic filter.

The NOG system at the ORR serves a variety of areas associated mainly with the reactor primary-water system and various experimental facilities (including air sweeps for valve boxes). Since the system was designed to remove gases from enclosures where high flow rates are not required, it is characterized by a relatively low flow (500 cfm) and a high negative pressure (-40 in. H_2O).

Since water is sometimes discharged into the NOG system during normal operation, a 35-gal catch tank and a water trap are provided at low points in the system's ducts to prevent accumulated liquids from blocking gas flow.

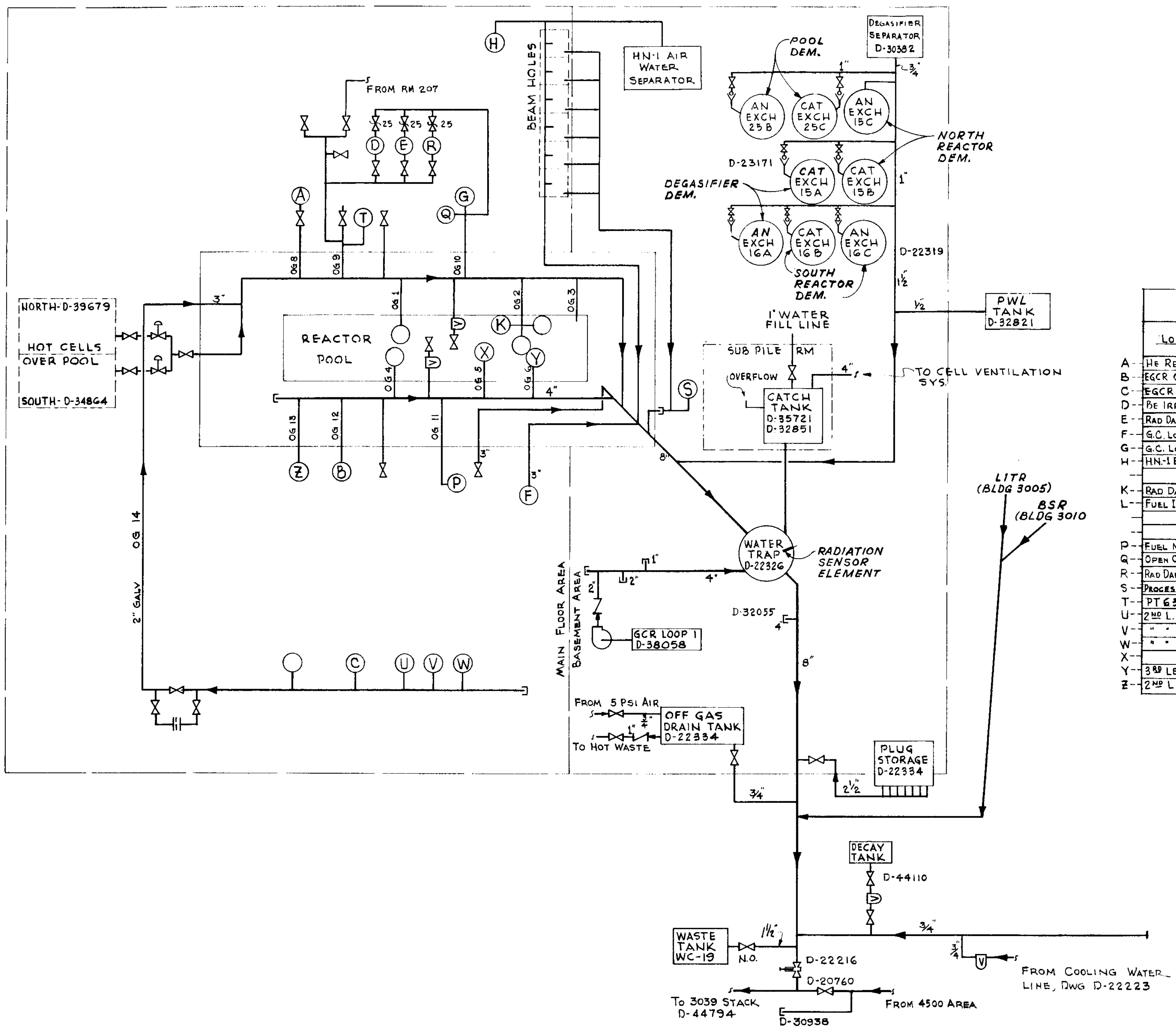
Numerous branches of the NOG system are located throughout the reactor building. The main duct for the NOG system in the ORR building is an 8-in. header extending from the third level to the basement area. The header is embedded in the southeast corner of the concrete structure of the storage-pool walls, and the main branches of the system are 4-in.-diam headers which surround the storage pools at the third-level elevation. Access ports are located in the reactor pool, around the outside of the pool parapet, at the east end of the reactor pool, and around the second-level balcony.

Connections to the NOG system are located inside the hot cells and vent small special enclosures within the cells. Also, they provide auxiliary ventilation capacity to ensure an adequate vacuum in the hot cells. When the cell-ventilation system cannot maintain an adequate vacuum in the cells, an automatic valve in the NOG line opens to maintain the correct vacuum.

Gases collected in the reactor tank are removed by the NOG system through an arrangement of liquid-gas separators and ball-float traps. In addition, beam-hole and experimental facilities are provided with NOG services. Numerous other facilities are provided with NOG connections, such as the degasifier, demineralizer columns, basement experiments, plug-storage holes, ^{16}N decay tank, and intermediate-level waste storage tank (WC-19).

4.5.2 Pressurizable Off-Gas (POG)

The NOG system was the only off-gas system originally in service at the ORR. As needs changed, long-term use of this system to satisfy the requirements of those experiments requiring off-gas service to operate auxiliary high-pressure systems raised two major objections. They were as follows: (1) Dependence was placed on positive-displacement units to maintain the off-gas capacity normally required (an electrical-motor-driven blower and an auxiliary steam-turbine-driven blower); and (2) some experiments were capable of accidentally releasing sufficient quantities of high-pressure gas to exceed the capacity of the NOG system, thereby pressurizing the entire NOG system, which serves other buildings in addition to the ORR complex. To alleviate this situation, the pressurizable off-gas system was designed and installed. The basic differences between the two off-gas systems are that there is



LEGEND

- MANUAL VALVE
- CENTRIFUGAL BLOWER
- AIR VENT
- CHECK VALVE
- MEASURING ORIFICE
- PNEUMATICALLY CONTROLLED VALVE
- PSI PRESSURE RELIEF VALVE W/ SETTING
- WATER FLOAT VALVE

NORMAL OFF-GAS USE

	Location	Applicable Drawings	(D)=Design, (M)=Measured and Date: Max CFM	Normal CFM	Neg Press In W.G.	Cleanup	Core Pos'n	Remarks
A	He Recoil							Not Currently Used
B	EGCR Capsules	D-44242				Charcoal Trap	PS 1,2,3,7,8,9	
C	EGCR Capsules	D-44242					PS 1,2,3,7,8,9	Valve Box System
D	Be Irr Study			O (D)			B 8	OG9 For Purge Service
E	Rad Damage Exp			O (D)			P6	OG10 For Pressure Relief
F	G.C. Loop 2	D-39315	47(M)10-62			ABS & Charcoal Can	South	Cell Press, Relief, Emer Vent
G	G.C. Loop 1	D-45186 & 7					B1-B2	
H	HN-1 Experiment	D-28055					—	HN-1 Existing But Not Used
K	Rad Damage Exp						P6	Not Currently Used
L	Fuel Irr Program						E9-F1	
P	Fuel Melt Down						F9	
Q	Open Cycle Loop						B9	
R	Rad Damage Study			O (D)			P5	See Remark For D & E
S	Process Inst Panel	(PT-64)			55(M)11-27-62		—	Pressure Transmitter
T	PT 63				52(M)11-27-62		—	Pressure Transmitter
U	2nd L. Balc (S)						F-2	Cubicle Ventilation
V	" " " (S)						C-1	" "
W	" " " "						F-3	" "
X							C-1	
Y	3rd Level						(Operations)	Pool Liner Evacuation
Z	2nd L. Balc (S)						C-1	Single Pass Fuel Loop

NOTE: DRAWING NUMBERS PERTAIN TO E & M DRAWINGS OF PHYSICAL SYSTEM INVOLVED.

Reference Drawings	Dwg. No.
Pressurizable Off-Gas System Schematic Diagram	D-44864
Cell Ventilation System - Schematic Diagram	D-44863

ORR SERVICES — BLDG NO. 3042

NORMAL OFF-GAS SYSTEM SCHEMATIC DIAGRAM

OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY
DIVISION OF UNION CARBIDE CORPORATION
OAK RIDGE, TENNESSEE

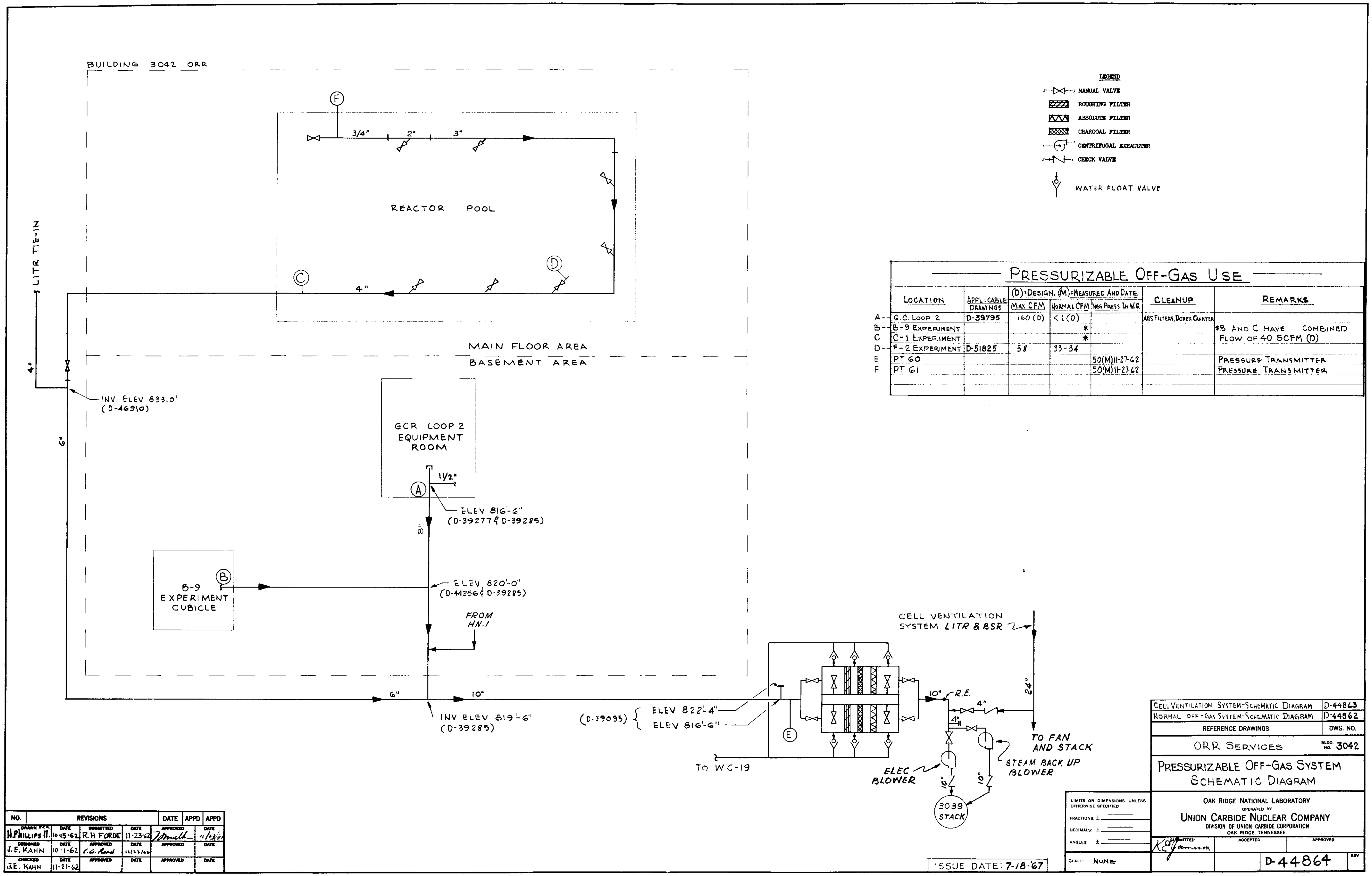
LIMITS ON DIMENSIONS UNLESS OTHERWISE SPECIFIED
FRACTIONS: ±
DECIMALS: ±
ANGLES: ±
SCALE: NONE

ISSUE DATE: 7-18-67

D-44862

NO.	REVISIONS	DATE	APPD	APPD
DRAWN: CLAPP/RAINES 10-16-62	SUBMITTED: R.H. FORDE 11-23-62	APPROVED	DATE: 1-23-63	
DESIGNED: J.E. KAHN 10-1-62	APPROVED	DATE: 11-23-62	APPROVED	DATE
CHECKED: J.E. KAHN 11-21-62	APPROVED	DATE	APPROVED	DATE

Fig. 4.3. Schematic Diagram of Normal Off-Gas System.



PRESSURIZABLE OFF-GAS USE

	Location	Applicable Drawings	(D)=Design, (M)=Measured and Date: Max CFM	Normal CFM	Neg Press In W.G.	Cleanup	Remarks
A	G.C. Loop 2	D-39795	160 (D)	< 1 (D)		Abs Filters, Dorex Canister	
B	B-9 Experiment			*			*B and C have combined
C	C-1 Experiment			*			flow of 40 SCFM (D)
D	F-2 Experiment	D-51825	38	33-34			
E	PT 60				50(M) 11-27-62		Pressure Transmitter
F	PT 61				50(M) 11-27-62		Pressure Transmitter

Fig. 4.4. Schematic Diagram of Pressurizable Off-Gas System.

a complete separation of air-sweep functions (as for valve boxes) and of pressurized-gas-discharge functions, and in the POG system a method is available for contained dissipation of pressurized gases in case of failure of any air-handling unit.

The pressurizable off-gas system was installed for use with experiments which will operate at a pressure greater than atmospheric and with a large air supply; that is, the system will provide a high-suction, low-flow exhaust of 500 scfm at -40 in. H_2O pressure. There is no concern if the gases discharged from experimental apparatus pressurize the off-gas system to as high as 100 psig if downstream stoppage occurs.

The system has several inlet points around the reactor pool for experiment tie-in. The main exhaust line for these access points begins at approximately the midsection of the reactor pool on the north side, continues eastward along the north wall of the reactor pool, southward across the east wall, westward along the south wall of the three pools, and penetrates the west pool wall at thimble T-23 on the second level. The line is shielded by 4 in. of lead from where it turns downward through the floor to the point where it penetrates the ORR building wall on the first level, east, approximately 5 ft underground. Just outside the ORR building the main line is joined by a 4-in. line, which provides service for the LITR. The main line then continues (as depicted in Fig. 4.4) toward the POG filter pit just east of Building 3089.

Prior to reaching the filter pit, the main line is joined by an 8-in. branch line at the south side of the ORR building. This branch line penetrates the south wall of the ORR building near the 24-in. cell-ventilation duct, basement level, just above the B-9 experiment cubicle. It then continues to the GCR Loop 2 equipment cell in the basement, through a charcoal trap and filter, and up and into the GCR Loop 2 containment cell.

After passing through the filters, the off-gas continues through a 10-in. overhead line to a 30-hp, 2000-cfm blower and is then discharged to the 3039 stack.

There is an interconnecting line between this system and the LITR cell-ventilation system to supply a sufficient volume of air to prevent the blower from overheating. A check valve and slide valve are installed in the line. The check valve prevents a reverse flow into the LITR cell-ventilation system, and the slide valve regulates the flow through the blower.

The filter pit (Fig. 4.5) consists of two banks of filtering systems (north and south). The filter banks may be used individually, in series, or in parallel, but normally either the north or the south bank is used individually.

Each filter bank is composed of four compartments. Within the first compartment, the inlet, there is a pair of valves for interconnecting the two banks and a roughing filter for removal of any particulate matter. The second compartment contains an activated-charcoal filter assembly for the removal of gaseous fission products, primarily iodine. The third compartment contains an absolute filter, and the last compartment contains the exit line and another pair of bank interconnection valves. The first, second, and fourth compartments have floor drains, with float traps, which discharge the condensate or water collected from the system to a hot drain line connected to waste tank WC-19.

An auxiliary 2000-cfm electric- or steam-powered blower has been connected to the POG system at the suction side of the main ORR pressurizable off-gas blower. Its purpose is to provide sufficient off-gas suction in case the ORR blower fails.

4.6 Exhaust Systems Instrumentation and Controls

The reliability of the exhaust systems is of utmost importance, even under abnormal conditions. To help ensure this reliability, sufficient instrumentation and control mechanisms have been provided.

4.6.1 Cell-Ventilation Instrumentation and Controls

Filter-Pit Temperatures. – The locations of the temperature sensing elements (thermocouples) in the filter pit are indicated in Fig. 4.6. The temperatures at these points are displayed on a multipoint recorder located in the ORR control room. An alarm at 38°C is effected through process annunciator "filter-pit high temperature."

Steam-Fan Controls (Automatic Startup). – Since operation of the cell-ventilation system is a requirement for ORR operation, the reactor power will be reduced automatically to 300 kw if the flow rate through the system should decrease to 2500 cfm. To ensure uninterrupted operation of the reactor, the steam-driven blower is instrumented

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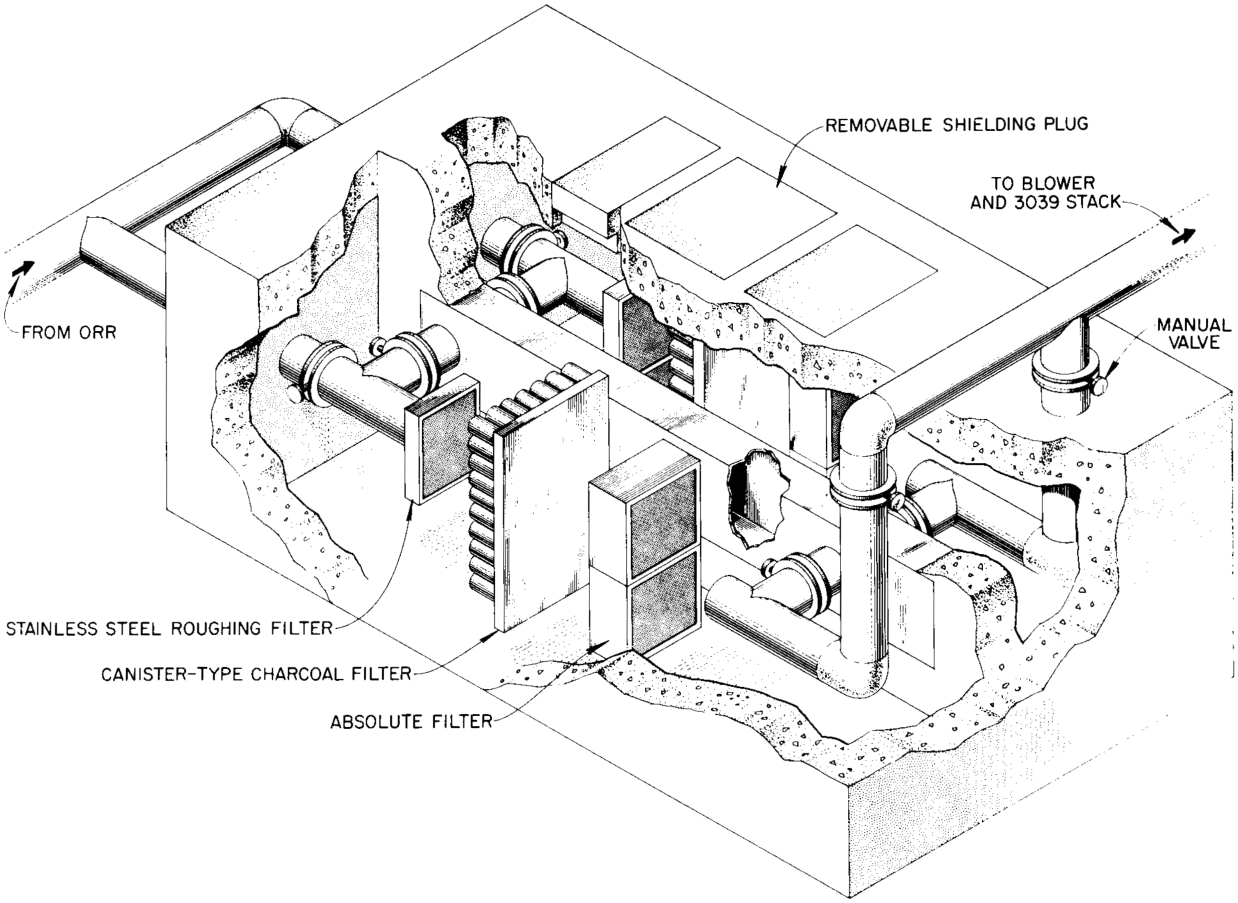

Fig. 4.5. POG Filter Pit.

to start automatically upon the failure of the electric blower.

Two virtually independent instrument channels are provided to effect the startup of the steam-driven blower upon an increase in the pressure in the cell-ventilation duct (indicating a failure of the electric blower or other trouble). One channel uses pneumatic instrumentation, the other electric. Both channels are designed to be "fail safe"; that is, loss of air pressure or electric power will result in a startup of the steam fan. This instrumentation is shown schematically in Fig. 4.7. These channels will hereafter be referred to as the pneumatic channel and the electric channel.

The pneumatic channel consists of a differential-pressure transmitter located at the duct over the B-9 cubicle, a pneumatic relay, and an air-operated control valve in the steam inlet line at the turbine.

Under normal operating conditions, plant air pressure (90 to 100 psig) is applied to the diaphragm of the pneumatically operated steam valve and to the air cylinder at the backdraft damper for the steam blower. This air cylinder compresses the assist-to-open spring, thus allowing the damper to close freely.

When the duct pressure increases to -2 in. H_2O, the compressed assist-to-open spring is released; consequently, the backdraft damper for the turbine opens as soon as the turbine blower provides sufficient suction.

The electrical channel consists of a differential-pressure switch, which will be actuated when the duct pressure increases to -2 in. H_2O, and a solenoid-operated block valve in the steam line.

Under normal operating conditions, the solenoid valve remains energized and therefore closed.

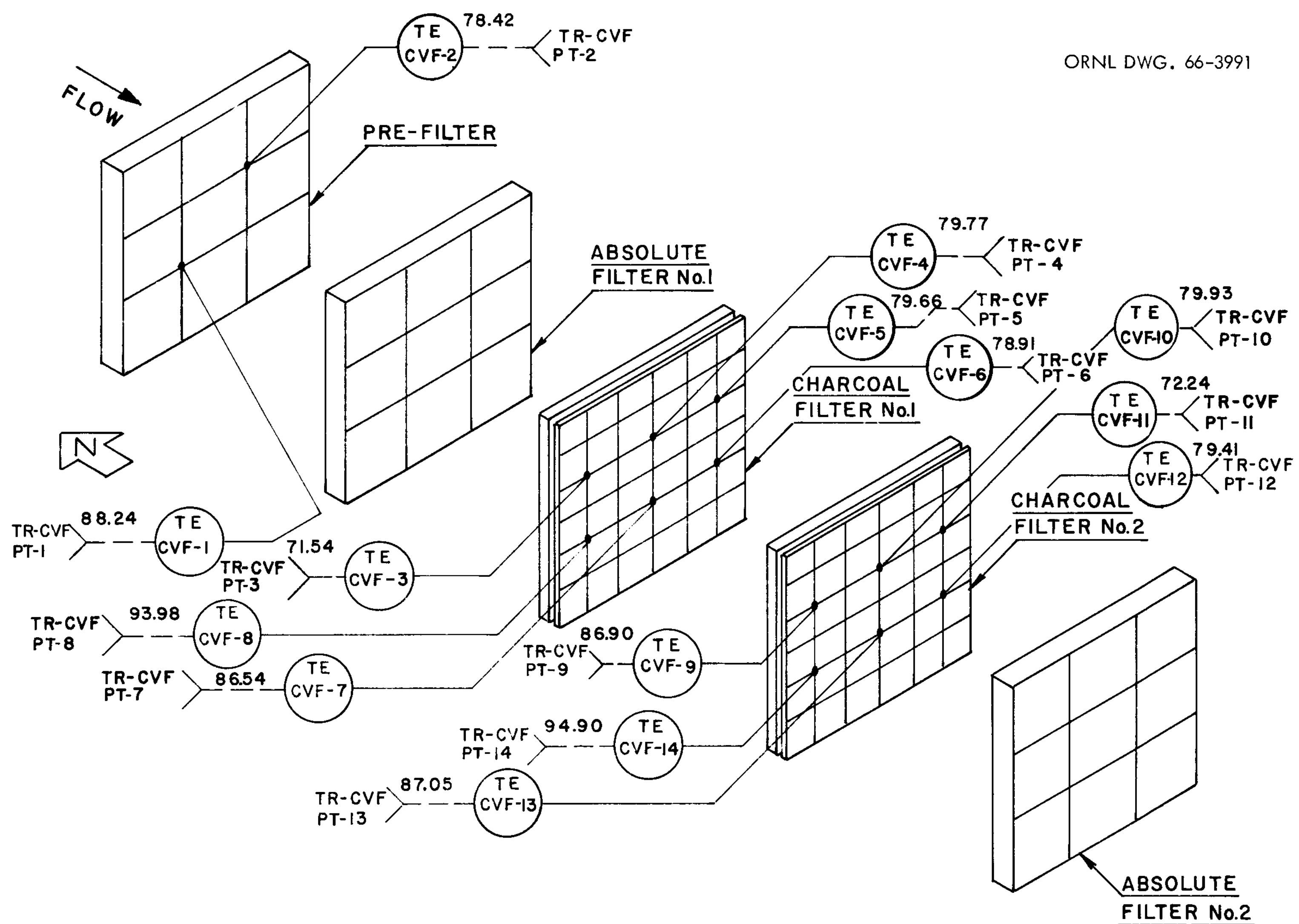

Fig. 4.6. Thermocouple Location in Cell-Ventilation Filter Pit.

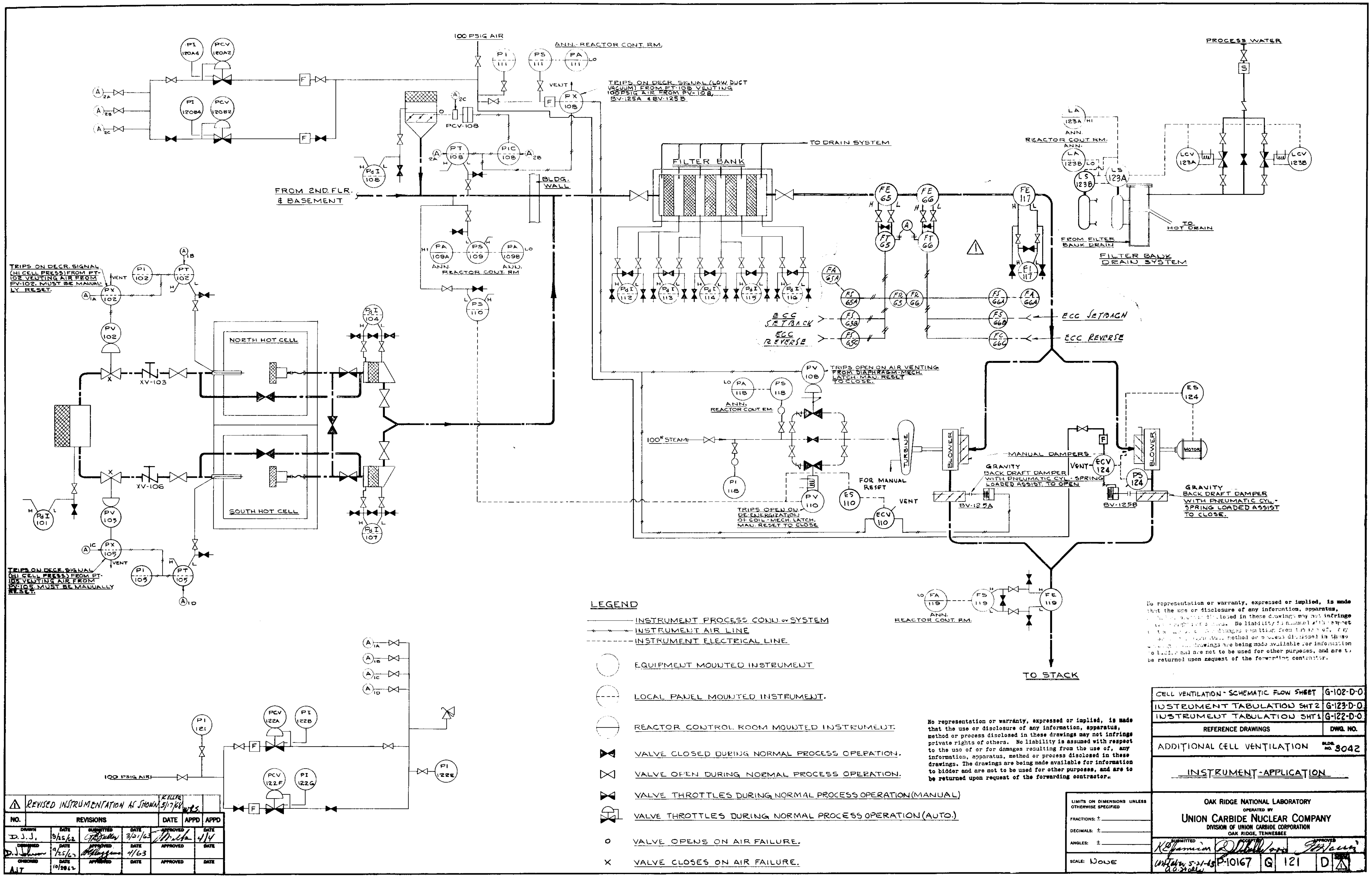

Fig. 4.7. Steam Fan Controls.

When the duct pressure increases to -2 in. H_2O, the differential-pressure switch is actuated and causes the solenoid-operated valve to be de-energized. This action, as well as a failure of electrical power, allows the valve to open and thereby supply steam to operate the turbine. As with the pneumatic channel, the compressed assist-to-open spring is released; consequently, the backdraft damper opens.

Main-Duct Pressure-Regulating Damper Controls. – The controls for the main-duct pressure-regulating damper are located in the basement. The differential-pressure transmitter, discussed in the steam-control section, sends a signal to the controller for the damper operator. This operator is reverse acting, so that the damper is wide open as the operator controller receives the 15-psig signal at low duct pressure and gradually closes as the signal is reduced from 15 psig, which corresponds to increasing duct pressure.

Level Controls for the Filter Drain and Seal Tank. – In the event of a low water level in the seal tank, float-type level switches, located in the underground pit with the tank, cause level-control valves to open. These valves are located in the pit on the two process-water supply lines. When they are open, water is added to the tank until the normal operating level is obtained; at that time the level switches initiate the closing of the level-control valves. Should the water level decrease further, a low-level alarm is initiated. If a high-level condition exists, an overflow line connected to the tank drains the excess water to waste tank WC-19. Should the water level rise above the overflow, the level switches initiate a high-water-level alarm.

4.6.2 NOG Instrumentation and Control

Instrumentation for the NOG system consists of pressure and radiation monitors. If an abnormal condition results in a loss of vacuum, the pressure-sensing instrumentation will initiate a reduction in reactor power. This instrumentation is dual-tracked, using two standard Foxboro dp cells as the sensing elements; each element is attached to the NOG system at a different location.

When the NOG pressure increases above -25 in. H_2O, an annunciator indicating NOG high pressure will alarm. If the NOG pressure increases further to a value above -20 in. H_2O, a reactor power reduction to 300 kw is effected through the reactor control system. If the NOG pressure increases to above -19 in. H_2O, an auxiliary reverse is initiated if within 5 sec the setback signal has not placed the reactor on a 100-sec negative period.

The instruments in the 3039 stack area, which monitor the status of components and operating conditions in that area, transmit their information to the Building 3105 control room. The parameters which are monitored include flow, pressure, and status of blower units. A pressure switch located in the inlet manifold of the blowers will activate the emergency steam-turbine-driven blower if the pressure increases to -27 in. H_2O. This switch is also activated upon failure of electrical power to the area.

The normal off-gas scrubber is equipped with an emergency steam-turbine-driven pump, which is activated upon the loss of pressure at the pump discharge or upon the loss of electrical power. Remote monitoring in Building 3105 indicates the status of the pumps, Δp across the scrubber, and Δp across the filters with an alarm to indicate high Δp.

The radiation level of the gaseous waste being discharged to the atmosphere from the plant-wide NOG system is monitored continuously. These data are recorded in Building 3105, and an alarm is initiated if the radiation level of the gases exceeds a predetermined value (15,000 counts/min). These local alarms are reported to the ORR shift engineer by the Tank Farm personnel.

At the WC-19 storage tank, a gage indicates the negative pressure maintained on WC-19 by the NOG system.

4.6.3 POG Instrumentation and Control

A pressure transmitter located at the filter pit is connected to the inlet line of the filter pit and sends its signal to a pressure recorder, located on a panel in the ORR control room. A pressure transmitter (located on the north balcony of the ORR) is connected to the $\frac{3}{4}$-in. pressurizable off-gas line located on the north side of the reactor pool and transmits its signal to a recorder, also located on the same panel in the ORR control room.

The same reactor control instrumentation is provided for the POG system as for the NOG, that is, alarm at -25 in. H_2O, setback at -20 in. H_2O, and reverse at -19 in. H_2O.

The differential-pressure instrumentation consists of two pressure gages. Four copper tubes are connected to each gage. The sampling points for these tubes are in each of the four filter compartments. By proper valving at the pressure-gage enclosure, located at the filter pit, the pressure drop across the absolute filter, the charcoal filter, or the roughing filter, or any combination of these, can be obtained.

An ionization chamber, located on the exit line from the filter pit, monitors the radioactivity level of the POG. The signal is transmitted to the pressurizable off-gas electrometer, located in the ORR control room, and, subsequently, to the POG radiation recorder, located on a panel also in the ORR control room. A recorder switch and a radiation alarm are actuated upon receiving a signal indicating twice the normal radiation background.

A transmitter, an integral part of the POG radiation recorder, relays the POG activity-level signal to a remote indicator, located on a panel in Building 3105; a recorder switch actuates a remote annunciator, also located on a panel in Building 3105. The purpose of these instruments is to alert the Laboratory's patrol chemical operators of the source of the high-level radioactivity which is being discharged into the 3039 stack, so that appropriate action can be taken.

5. REACTOR AND EXPERIMENTAL FACILITIES

5.1 Introduction

The ORR uses MTR-type enriched-uranium fuel elements and beryllium reflector elements in a seven- by nine-element rectangular lattice, with neutron moderation and core cooling provided by demineralized light water. The reactor vessel is submerged in one end of a water-filled rectangular pool. Control-rod drives are operated from below the reactor, as shown in Fig. 1.3. The reactor is housed in a four-storied building about 100 by 108 ft. A 6-ft-wide balcony extends around the outside of the pool structure at an elevation roughly midway between the first- and second-floor elevations, as shown in Fig. 5.1.

The ORR characteristics of greatest importance to experimenters are as follows:

Operating power: 30 Mw

Thermal-neutron flux: 1.4×10^{14} neutrons cm^{-2} sec^{-1} (average)
5×10^{14} neutrons cm^{-2} sec^{-1} (maximum)

Neutron-flux distribution: see ref. 1 and Sect. 7

Gamma heating: 3 to 10 w/g depending on material and lattice position

Coolant temperature: reactor cooling-water inlet temperature, 120°F

Coolant pressures: ~11 psig below core
36 psig above core (full flow)

Operating cycle: 8 weeks, comprising 7 weeks at full power and one week of shutdown for maintenance, experiment changes, and refueling

Refueling cycle: ~12 days; within each operating cycle there are three short (~½ day) shutdowns for refueling

Operating time: ~80%

Unscheduled shutdown frequency: average of about one per month

Operating costs: $80,000 per month (includes labor, overhead, materials, and fuel fabrication cost, but no ^{235}U cost, reprocessing cost, or depreciation)

5.2 Core and Reflector Components

The reactor core and reflector components, which include the fuel elements, shim rods, beryllium reflector pieces or elements, isotope production facilities, and in-tank experimental facilities, are all located in the support grid which supports the core. The support grid and the core and beryllium reflector components are located entirely within the reactor vessel. No upper grid is used in the ORR; instead, hold-down "arms" are used, which serve the same purpose as an upper grid. These hold-down arms (Fig. 5.2) contain the upper guide bearings for the shim rods and span columns 4 and 6 (Fig. 5.9) of the core support grid to align and firmly secure the fuel elements around the shim rods.

[1]C. D. Cagle and R. A. Costner, Jr., *Initial Post-Neutron Measurements in the ORR*, ORNL-2559 (May 28, 1959).

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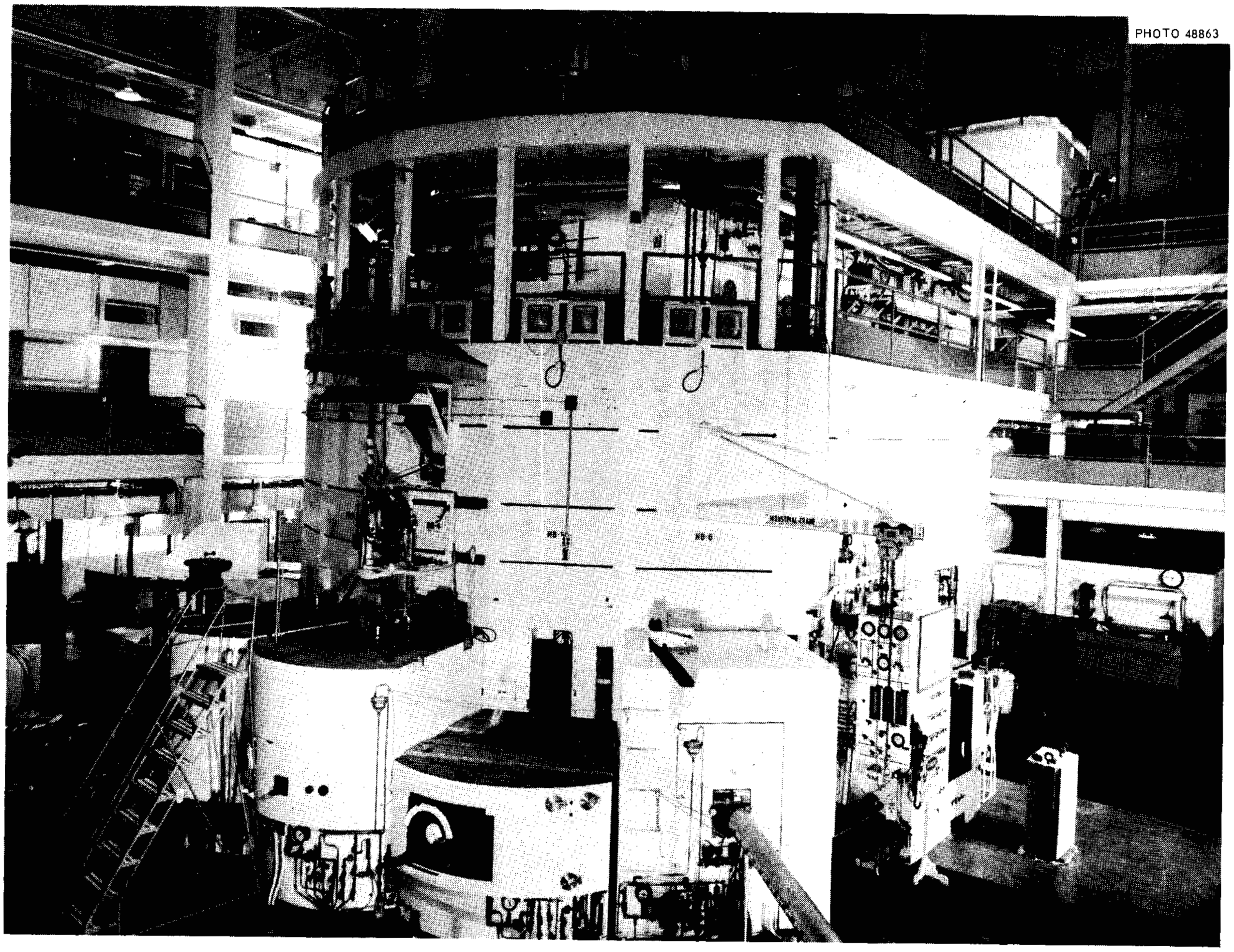

Fig. 5.1. Beam-Hole Experimental Area of the ORR.

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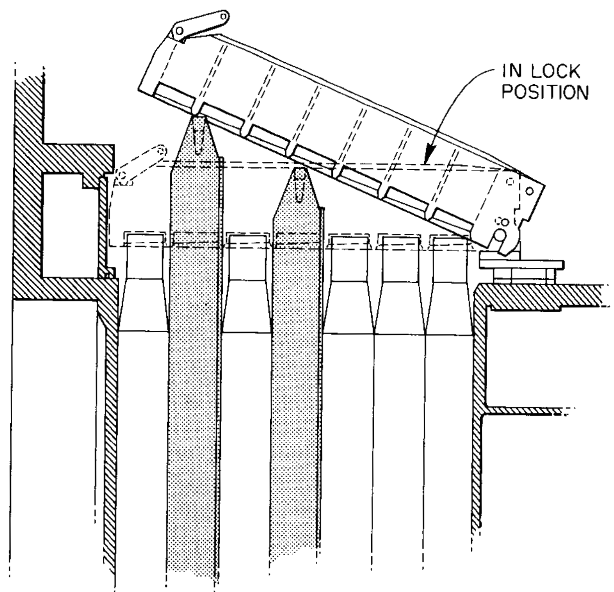

Fig. 5.2. Hold-Down Arm.

5.2.1 Fuel Elements

The fuel elements used in the ORR are of the aluminum-clad-plate-type design (Figs. 5.3 and 5.4) and now contain a total of 240 g of ^{235}U when new. See Appendix B for complete fuel element specifications. Each fuel element is made of 19 composite plates containing the ^{235}U fuel in the form of an aluminum-clad uranium-aluminum alloy. The fuel-element end boxes were designed to minimize the pressure drop across the core and are identical to permit the elements to be inverted if desired.

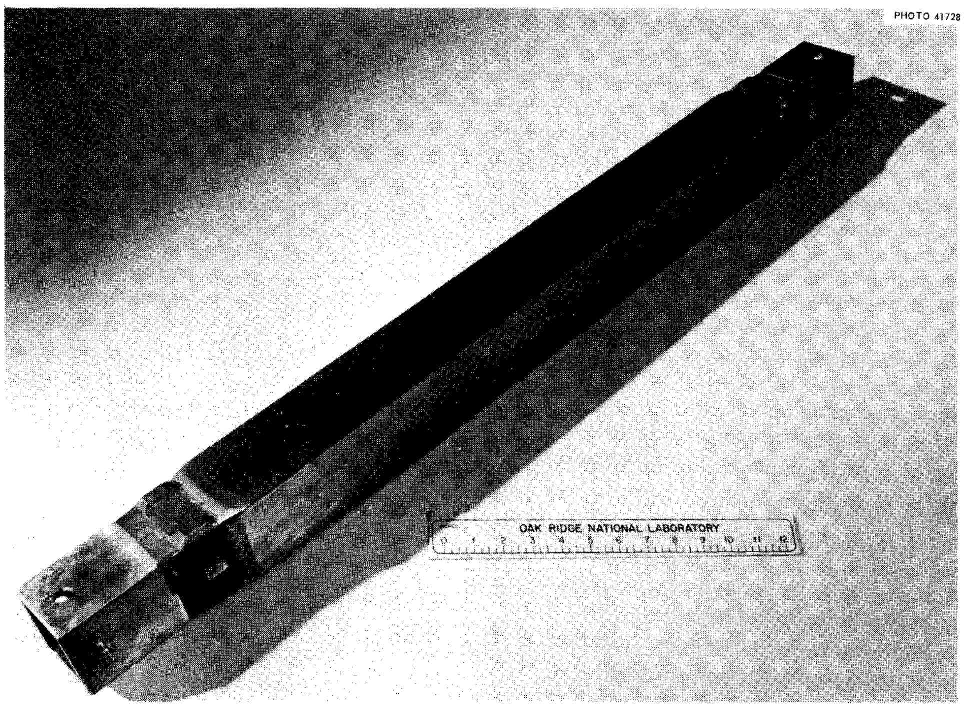

Fig. 5.3. Fuel Element.

Fig. 5.4. End View of Fuel Element.

Table 5.1. Dimensions of ORR Fuel Elements

Unit	Nominal Dimension[a]
Element assembly	
Length	$38\frac{3}{8}$
Width (through the side plates)	2.996
Width (center line through outside fuel plates)	3.068
Plate spacing	0.104 (min)
Inside fuel plates	
Thickness (overall)	0.050
Length (overall)	$24\frac{5}{8}$
Clad thickness	0.015
Core (alloy) thickness	0.020
Core (alloy) length	$23\frac{5}{8}$
Core (alloy) width	2.5
Width (before bending)	2.8
Outside fuel plates	
Thickness (overall)	0.065
Length (overall)	$27\frac{1}{8}$
Clad thickness	0.0225
Core (alloy) thickness	0.020
Core (alloy) length	$23\frac{5}{8}$
Core (alloy) width	2.5
Width (before bending)	2.8
End box locating pads	
Distance (through flat sides)	3.032–3.034
Distance (concave to convex side)	3.186–3.188
Miscellaneous	
Effective fuel length	23.625
Effective volume	228.4 in.^{3}
Metal volume	89.59 in.^{3}
H_2O volume	138.8 in.^{3}
Heat transfer area	16.58 ft^2

[a]In inches unless otherwise indicated.

Table 5.2. Weights of ORR Fuel Elements

Description of Components		Nominal Weight (kg)
Nineteen fuel plates		3.30
Alloy cores (21.7 wt % U, balance Al)	1.19	
Cladding (aluminum)		
Thirty-four 0.015-in.-thick sections	1.77	
Four 0.0225-in.-thick sections	0.34	
	3.30	
Two side plates (aluminum)		1.15
Two combs (aluminum)		0.01
		4.46

The individual fuel plates are of a sandwich-type construction and are composed of a fuel-containing alloy completely clad with aluminum. The cladding is metallurgically bonded to the fuel alloy. The fuel-containing alloy is a mixture of uranium and aluminum containing ~21.7 wt % uranium with the remainder being aluminum.

Uranium used in the alloy contains a minimum of 93 wt % ^{235}U, and the aluminum is of a high purity grade. Each of the 19 fuel plates contains 12.63 g ± 4% of ^{235}U, and each fuel element contains a total of 240 g ± 2% of ^{235}U. Cladding of thickness 0.015 in. and an alloy thickness of 0.020 in. make up the total 0.050-in. thickness of the inner fuel plates of an element. The alloy core has a width of 2.5 in., thus providing 0.15 in. of aluminum on each edge of the fuel plate, and the active length of a fuel plate is $23\frac{5}{8}$ in. The two outside fuel plates of an element are similar to the inner ones except that the cladding thickness is 0.0225 in., giving a total plate thickness of 0.065 in. Before assembling into elements, the 2.8-in.-wide plates are curved on a $5\frac{1}{2}$-in. radius. Characteristics of the fuel elements are given in Tables 5.1 through 5.3. The pertinent thermal characteristics of the elements are given in Sect. 7.5.

5.2.2 Shim Rods

Two types of shim rods are used at the ORR. One type has a cadmium poison section with a fuel follower. The other type has the same kind of poison section, but the follower is made of aluminum. The first type is normally used in the fuel region of the core and the latter in the reflector region. Spent shim rods of the first type may be used to replace the aluminum-follower rods, however, since the remaining fuel content is small (see Sect. 7.3.6).

Table 5.3. Composition of ORR Fuel Elements[a]

Material	Volume (cm^3)	Volume Fraction	Weight[b] (g)	Apparent Density (g/cm^3)	Atomic or Molecular Density (atoms or molecules/cm^3)
Uranium	13.67	0.003652	257	0.06866	1.759×10^{20}
^{235}U	12.77	0.003412	240	0.06412	1.6434×10^{20}
^{238}U	0.90	0.000240	17	0.004542	1.1494×10^{19}
Aluminum	1454	0.3885	3926	1.0516	2.3476×10^{22}
H_2O	2275	0.6078	2271	0.6067	2.0301×10^{22}

[a]T. P. Hamrick, *ORR Fuel Units – Material Composition, Cross Sections, and Calculation of* $k\infty$, ORNL-CF-64-10-19 (Oct. 1, 1964).

[b]Weight of effective length only.

Each shim rod consists of three main sections: the upper or cadmium poison section, the middle or follower section, and the lower or piston section.

The cadmium and piston sections of each of the two types of shim rods are identical; the only differences are in the followers.

Poison Section. – The cadmium section consists of a 0.040-in.-thick sheet of cadmium which is clad on both sides with 0.020 in. of aluminum. This composite sheet is formed into a hollow square box-like structure 2.345 in. square by $\sim 30\frac{1}{2}$ in. long. This cadmium-box insert is contained in an aluminum shell whose outer dimensions are identical to those of the follower section and whose nominal wall thickness is 0.234 in. There is no size transition between the follower section and the poison section; instead, the outer parts of the sections form one continuous profile (see Figs. 5.5 and 5.6).

The general characteristics of the shim rods are given in Table 5.4.

Table 5.4. Characteristics of ORR Shim Rods

Unit	Nominal Dimension (in.)
Shim rod assembly	
Total length (poison, follower, and piston)	$117\frac{11}{16}$
Width (through flat sides)	2.838
Width (center line through curved sides)	3.029
Center line cadmium to center line follower	$\sim 29\frac{1}{4}$
Bottom of cadmium to top of fuel in follower	~ 2
Upper poison section, jacketed cadmium insert	
Length (overall)	$30\frac{3}{4}$
Length (cadmium)	$30\frac{1}{2}$
Outside dimensions (width)	2.345
Insert thickness	0.080
Cadmium thickness	0.040
Jacket thickness	0.020

Follower Sections. – The shim-rod follower that contains fuel is very similar in construction to the ORR fuel elements. They are of the aluminum-clad-plate-type design and contain a total of 154 g of ^{235}U. Appendix B gives complete specifications for the fuel followers. A summary of the specifications is given in Tables 5.5, 5.6, and 5.7. Each follower contains 14 fuel-bearing plates besides the two outer curved solid aluminum plates that form part of the outer housing.

The fuel plates are of the sandwich-type construction and are somewhat thicker than the fuel plates used in the fuel elements. The uranium-aluminum alloy fuel core (19.5 wt % uranium) is 0.020 in. thick and is clad with a 0.020-in.-thick layer of aluminum to form a 0.060-in.-thick plate. Each plate contains 11.0 g $\pm 4\%$ of ^{235}U. Fuel plates are formed with a $5\frac{1}{2}$-in. radius before they are assembled into the follower section.

The aluminum shim-rod follower is a block of aluminum whose outer dimensions are the same as those of a fuel follower. A $\frac{1}{2}$-in. hole drilled

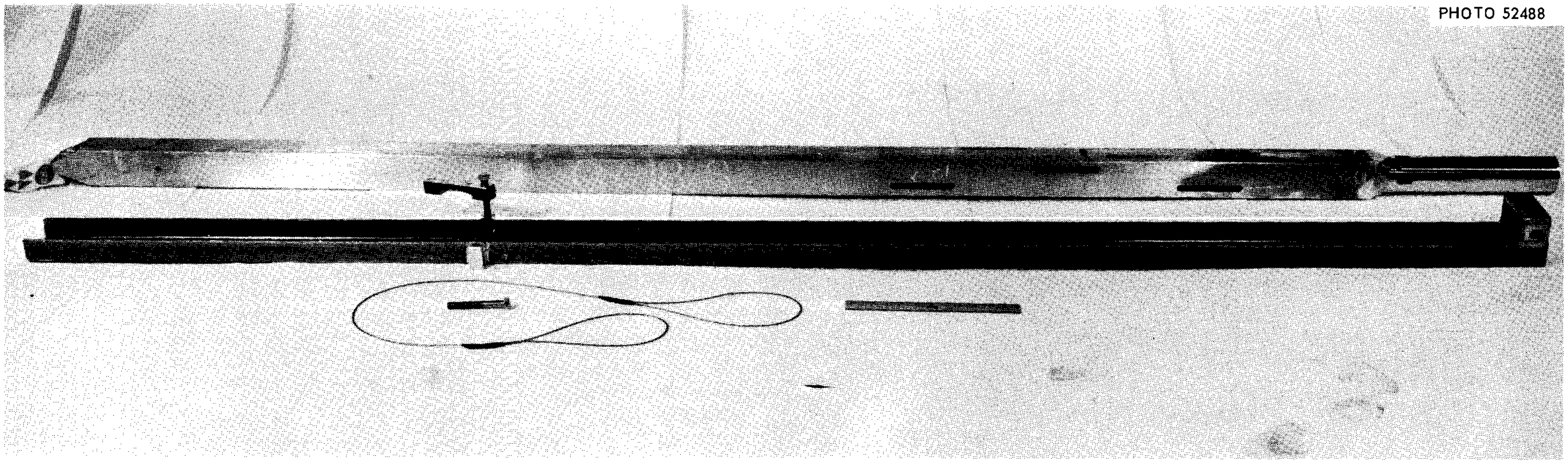

Fig. 5.5. Shim Rod.

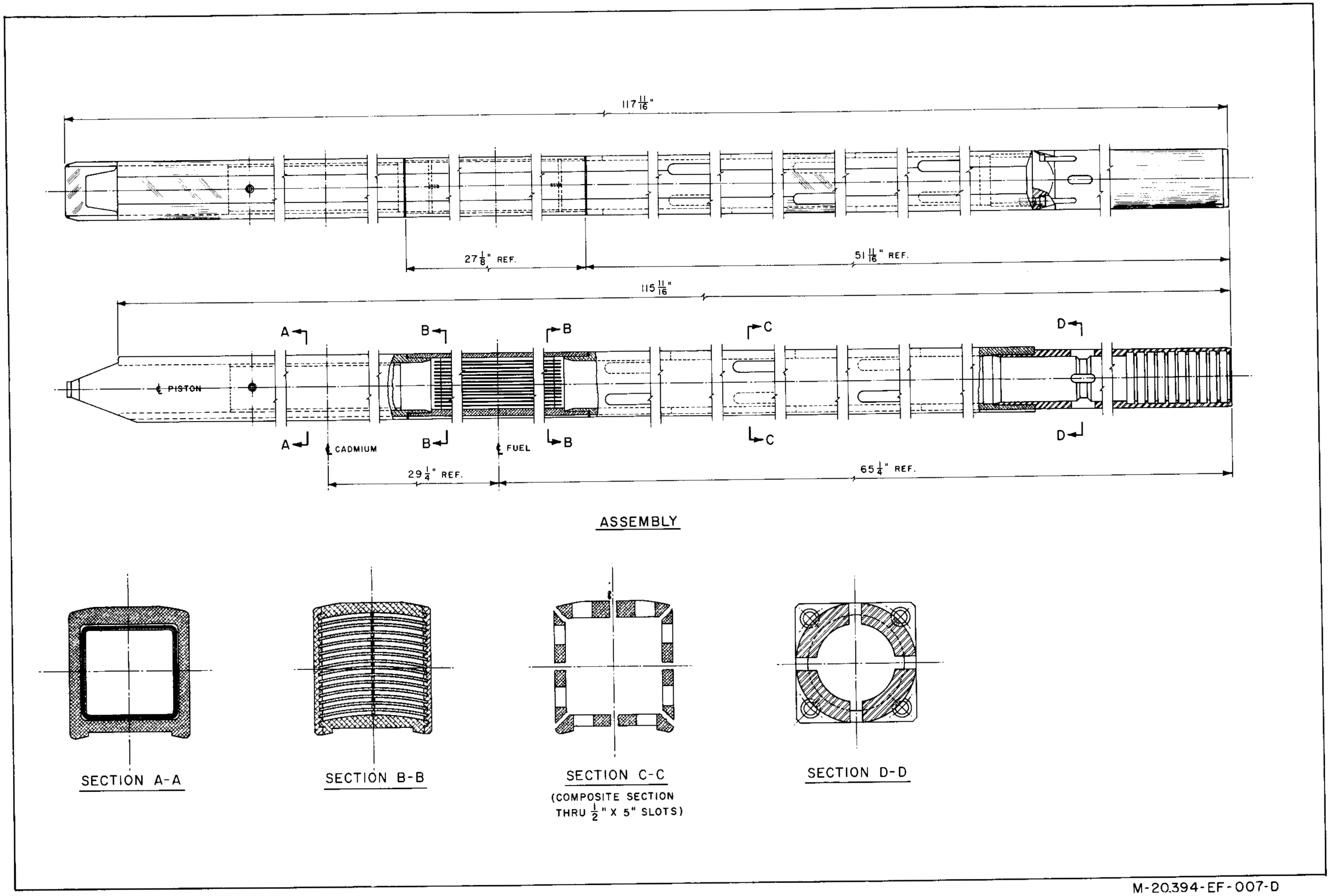

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Fig. 5.6. Shim-Rod Assembly.

Table 5.5. Characteristics of ORR Shim-Rod Fuel Followers
154 g, 14 plates

Unit	Nominal Dimension[a]
Fourteen-plate fuel follower section	
Plate spacing	0.117 (typical)
Plate thickness (overall)	0.060
Plate length (overall)	$24\frac{5}{8}$
Clad thickness	0.020
Core (alloy) thickness	0.020
Core (alloy) length	$23\frac{5}{8}$
Core (alloy) width	2.5
Plate width (before bending)	2.8
Miscellaneous	
Effective fuel length	23.625
Effective volume	228.4 $in.^3$
Metal volume	122.6 $in.^3$
H_2O volume	105.8 $in.^3$
Heat transfer area	12.22 ft^2

[a]In inches unless otherwise indicated.

Table 5.6. Weights of ORR Shim-Rod Fuel Section

Component		Nominal Weight (kg)
Fourteen fuel plates		2.63
Alloy cores (19.5 wt % uranium, balance aluminum)	0.85	
Cladding (aluminum)	1.78	
	2.63	
Heel and toe plates (curved sides) (aluminum)		1.50
Two side plates (aluminum)		1.14
Two combs (aluminum)		0.01
		5.28

Table 5.7. Composition of ORR Shim-Rod Fuel Followers

Material	Volume (cm^3)	Volume Fraction	Weight[a] (g)	Apparent Density (g/cm^3)	Atomic or Molecular Density (atoms or molecules/cm^3)
Uranium	8.8	0.002351	166	0.04435	1.1367×10^{20}
^{235}U	8.2	0.002190	154	0.04114	1.0544×10^{20}
^{238}U	0.6	0.000160	12	0.003205	8.1119×10^{12}
Aluminum	2000	0.5343	5400	1.4427	3.2201×10^{22}
H_2O	1734	0.4633	1731	0.4625	1.5476×10^{22}

[a]Weight of effective length only.

longitudinally along the center line of the follower allows a flow of coolant water through the aluminum, and the longitudinal dimensions are such that the overall length of the shim rod with an aluminum follower is the same as that of one with a fuel follower.

Piston Section. – The stainless steel shock-absorber piston section of a shim rod is simply a transition region that changes from the overall outer dimensions of the follower section to a cylindrical section with a 2.437-in. outer diameter and a $2\frac{1}{4}$-in. inner diameter. When the shim rods fall due to a reactor scram, the piston sections drop into shock-absorber cylinders to cushion the impact of the fall. The shock-absorber cylinders are anchored to the bottom head of the reactor tank. The lower cylindrical sections of the shim rods are provided with inside circumferential grooves which are engaged by the latching mechanisms of the rod drives.

5.2.3 Beryllium Reflector Elements

The beryllium reflector elements are designed so that they will fit into all core grid positions and may be interchanged with fuel elements and experimental facilities. The outside appearance of a reflector element is very similar to that of a fuel element. Aluminum end boxes similar to those used on fuel elements are fastened onto both ends of the beryllium portion and allow the element to be inverted if desired. The beryllium block has the same overall dimensions as a fuel element (Fig. 5.4). Overall dimensions are 3.010 in. from flat side to flat side and 3.320 in. from the most extreme point on the convex side to the "flats" on the concave side. The overall length is the same as the length of a fuel element.

Two types of beryllium elements are in use. Both have the same outer dimensions and end box arrangements, but they differ regarding the internal construction of the beryllium portion. One type is a block of beryllium with a $\frac{3}{16}$-in. coolant hole drilled longitudinally through the center of the block. This type is used only as a reflector component.

The other type is designed to be used either as a reflector element or as an experimental facility. It consists of a beryllium block with a 2-in.-diam hole drilled through its center and a $1\frac{7}{8}$-in.-diam solid beryllium insert or plug that may be inserted into the hole. Without the insert in place, the element may conveniently be used as an experimental facility which provides a predominantly thermal neutron flux; with the beryllium insert in place, it may be used as a normal reflector element.

5.2.4 Isotope Production Facilities

Isotope production facilities, sometimes called isotope stringers, are used to contain samples of materials being irradiated in the high neutron flux in the reflector region for the production of radioactive isotopes. These stringers are placed in the hollow beryllium pieces for irradiation and are sized to fit inside the hole and still provide a coolant gap around its outside. One of these isotope stringers is shown in the foreground of Fig. 5.7. The isotope stringers consist of ten small cylindrical sections called trays stacked end to end to form a long cylindrical section. The trays are $2\frac{1}{8}$ in. long by $1\frac{3}{4}$ in. in outside diameter and are assembled on $3\frac{1}{16}$-in. spaces along the piece to give a total length of ~ 33 in., including the end sections. The aluminum trays are assembled on a $\frac{3}{8}$-in.-diam stainless steel rod,

Fig. 5.7. Isotope Production Facility.

which spaces the trays longitudinally and also serves as a pivot whereby each tray may be rotated outward for loading or repositioning the samples. The rod fits through one of the circumferential holes in each tray.

There are two types of trays; one type accommodates six $\frac{1}{2}$-in.-OD capsules, and the other accommodates eight $\frac{3}{8}$-in.-OD capsules. The trays may be intermixed in a stringer, or they may be all of one size. In both types of trays the positions of all but one of the capsules are located in a circle near the outside of the tray, and the other is located in the center.

In addition to the holes where the capsules fit, there are eight $\frac{3}{16}$-in.-diam holes drilled on a circle between the central position and the circumferential positions. The holes allow coolant to flow downward around the capsules and outward to the coolant gap around the outside of the stringer through sixteen $\frac{7}{16}$-in.-diam holes drilled into the cylindrical surface of each tray. The holes also decrease the metal-to-water ratio in the assembly.

Samples to be irradiated in the stringers are sealed in aluminum capsules $2\frac{1}{2}$ in. long and either $\frac{1}{2}$ or $\frac{3}{8}$ in. in outside diameter. The capsules containing materials to be irradiated are welded closed and leak tested prior to installation in a sample tray.

5.2.5 In-Reactor Experimental Facilities

The In-Reactor Experimental Facilities provide the highest neutron fluxes (up to 5×10^{14} neutrons cm^{-2} sec^{-1}) available in the reactor. Access is through flanges in the reactor tank top, but the reactor can be refueled without disturbing experiments (Fig. 5.8). A typical, but not current, allocation of in-reactor irradiation positions and access flanges to experiments is shown in Fig. 5.9. In-reactor grid positions are identified by a combination of a letter (*A* through G) and a numeral (1 through 9).

5.2.6 Hydraulic Tube Systems

System No. 1. – The No. 1 hydraulic tube system, Fig. 5.10, provides a relatively simple means for inserting and removing capsules (targets for radioisotope production) while the reactor is operating. It will accommodate a total of 12 individual target capsules, each of which might contain several isotopes. Capsules are scheduled for irradiation by research, Isotopes, and Operations personnel, depending upon circumstances which vary from day to day. All capsules, however, must meet size and weight specifications, must be properly packaged, and must be approved by the Operations Division Technical Assistance Department.

The system is designed to use water to push capsules into and out of the reactor core position F-8. Water from the pool coolant pump or from the reactor cooling system, as required, supplies the necessary flow and pressure for inserting, removing, and cooling the capsules. Valves at the loading stations allow the water flow to be directed toward the reactor, carrying the capsule into the core. The water is then discharged through a connecting tube to the decay tank and sent back into either the pool system or the reactor water system. For removing the capsule, part or all of the water flow through the other hydraulic tubes is directed back up the hydraulic tube from which the capsule is to be removed, carrying the capsule back to the loading station.

Normally, water is supplied from the pool coolant pump and discharged to the reactor pool. Five other supply-discharge valving arrangements are possible; however, three of these are undesirable because of the interchange of water between the reactor and pool systems. The six possible valving arrangements are listed below as normal, special, and undesirable. Special valving arrangements are to be made only upon receipt of written instructions from the reactor supervisor. Undesirable valving arrangements are strictly avoided.

1. Normal – water is supplied from the pool coolant pump and discharged to the reactor pool.
2. Special – water is supplied from the pool coolant pump and is discharged to the pool cooling system through the degasifier.
3. Special – water is supplied from the reactor cooling system and is discharged into the reactor cooling system through the degasifier.
4. Undesirable – water is supplied from the pool coolant pump and discharged into the reactor system through the degasifier. Note: This arrangement would result in a flow from the reactor tank to the pool through the equalizer leg.
5. Undesirable – water is supplied from the reactor cooling system and is discharged to the reactor pool.

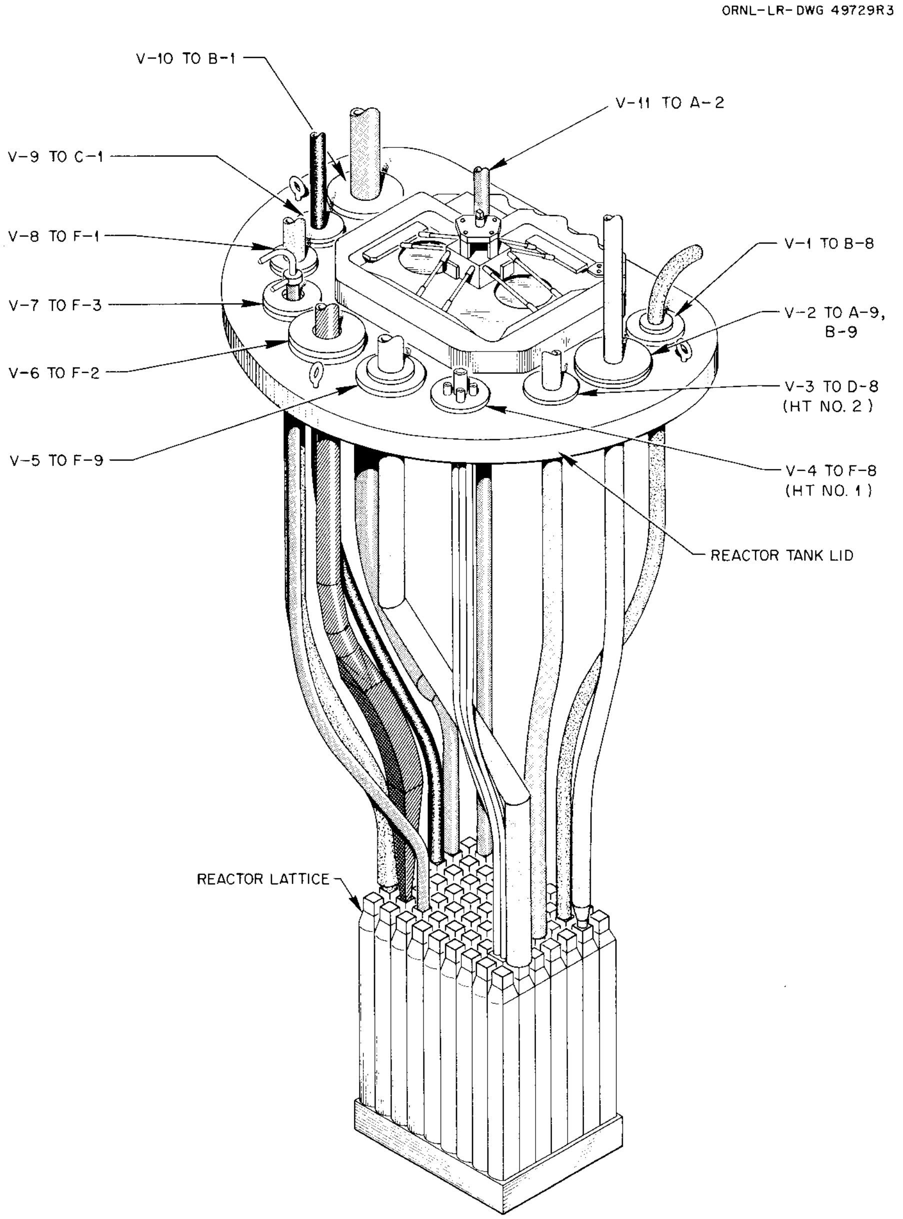

Fig. 5.8. In-Reactor Experimental Facilities.

6. Undesirable – water is supplied from the reactor cooling system and is discharged into the pool cooling system through the degasifier.

System No. 2. – The No. 2 hydraulic tube system for the ORR, Fig. 5.11, is a facility designed to allow irradiation of as many as 25 capsules simultaneously: 20 capsules $\frac{1}{2}$ in. OD by $2\frac{1}{2}$ in. long and 5 capsules $\frac{5}{8}$ in. OD by $2\frac{1}{2}$ in. long. The facility consists of five separate tubes that are essentially independent of each other. Tubes 21, 22, 23, and 24 accommodate five $\frac{1}{2}$-in.-OD capsules each, and tube 25 accommodates five $\frac{5}{8}$-in.-OD capsules.

The loading station, located on the north wall of the center pool, consists of five loading chambers, each of which is provided with a hinged top and lockdown mechanism to facilitate insertion and removal of capsules. Each loading chamber is designed with a water control valve built as an integral part of the loading chamber. This valve

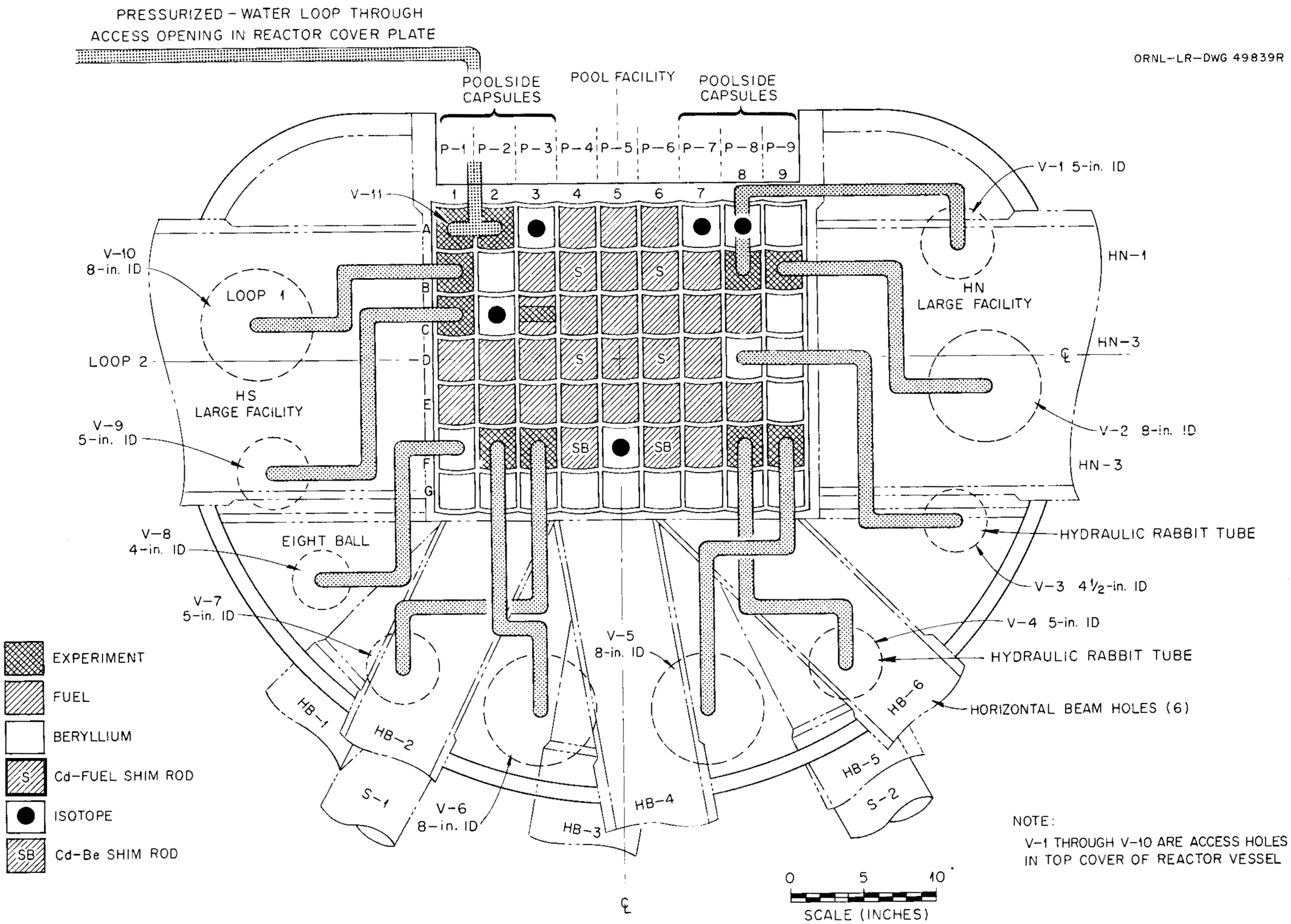

Fig. 5.9. Lattice Pattern and Experiment Locations.

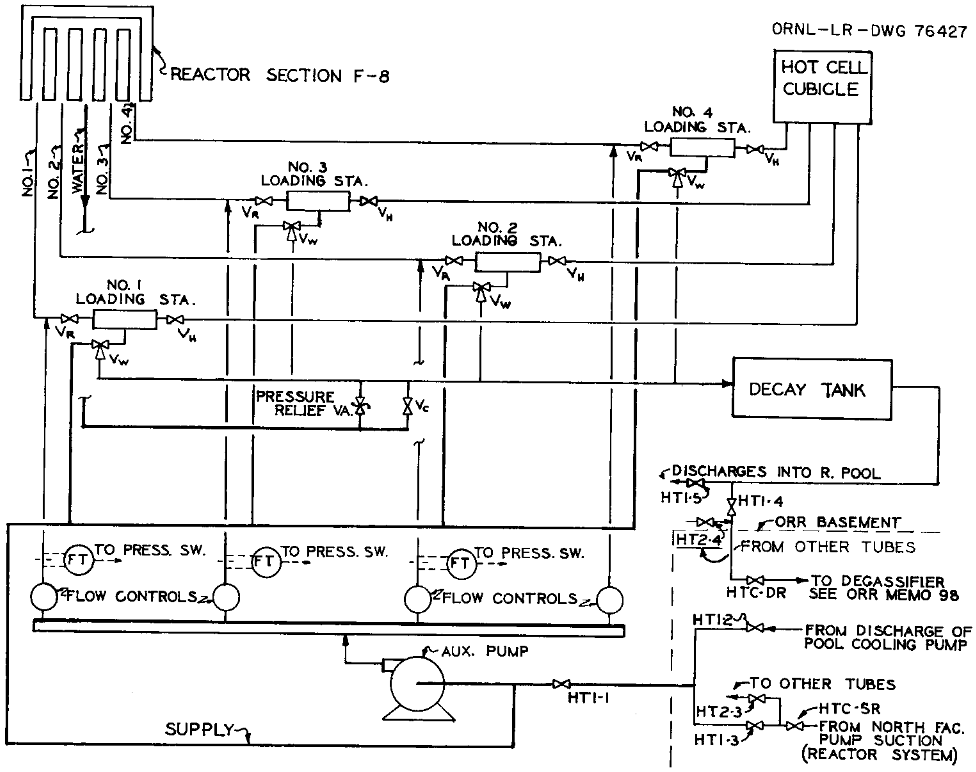

Fig. 5.10. Schematic Diagram of No. 1 Hydraulic Tube System.

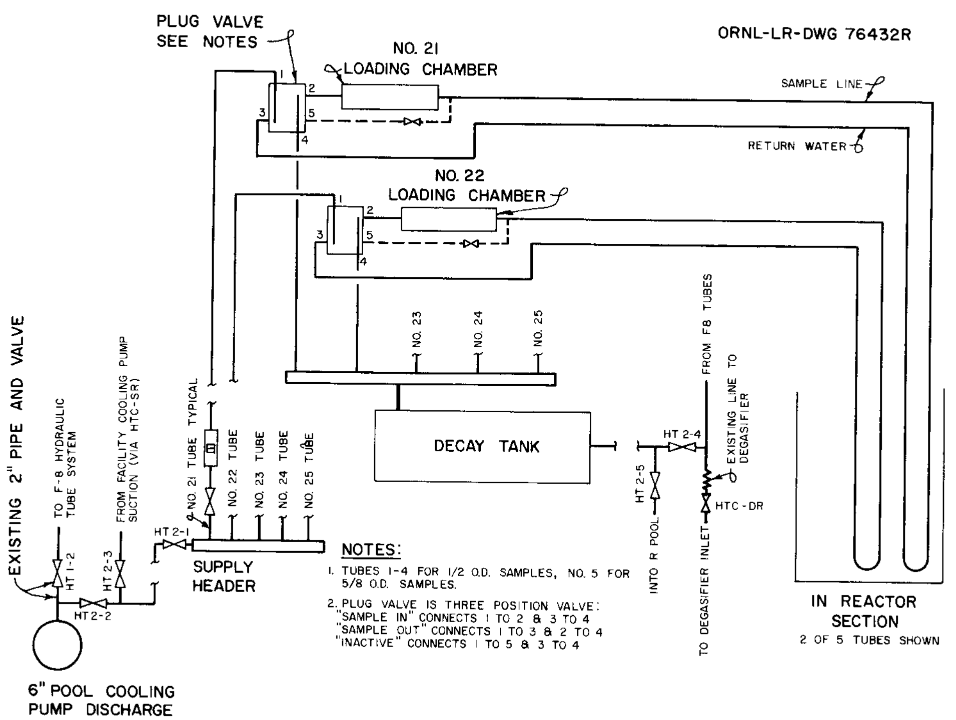

Fig. 5.11. Schematic Diagram of No. 2 Hydraulic Tube System.

directs the water flow that moves the capsules to or from the reactor. The water control valve is a specially designed duplex valve with one control stem. There are three operating positions for each valve, as described below. The water control valve determines three system conditions.

The system is designed to use water to push capsules into and out of the reactor grid position D-8. Water from the pool coolant pump or from the reactor cooling system, as required, supplies the necessary flow and pressure for inserting, removing, and cooling the capsules. Valves at the loading stations allow the water flow to be directed toward the reactor, carrying the samples into the reactor. The water is then discharged from the reactor through a separate line back through the four-way valve, where it forms a common return with the other four tubes. It is then sent through the decay tank and back into either the pool water system or the reactor water system. For removing the capsule, the supply water is directed toward the reactor through the return line. This forces the capsule out of the reactor and back into the loading chamber. Normally, water is supplied from the pool coolant pump and discharged to the reactor pool. Five other supply-discharge valving arrangements are listed below as normal, special, and undesirable. Special valving arrangements are to be made only upon receipt of written instructions from the reactor supervisor. Undesirable valving arrangements are strictly avoided.

1. Normal – water is supplied from the pool coolant pump and discharged to the reactor pool.
2. Special – water is supplied from the pool coolant pump and discharged to the pool cooling system through the degasifier.
3. Special – water is supplied from the reactor cooling system and is discharged into the reactor cooling system through the degasifier.
4. Undesirable – water is supplied from the pool coolant pump and discharged into the reactor system through the degasifier. Note: This arrangement would result in a flow from the reactor tank to the pool through the equalizer leg.
5. Undesirable – water is supplied from the reactor cooling system and is discharged to the reactor pool.
6. Undesirable – water is supplied from the reactor cooling system and discharged into the pool cooling system through the degasifier.

5.3 Reactor Vessel

The reactor vessel serves as the containment system for the reactor core and for the primary coolant system while it passes through the core region. The vessel is irregular in cross-sectional shape, but in general it somewhat resembles a right circular cylinder that is partially flattened on one side (Fig. 1.3). The tank is ~18 ft tall and is made of aluminum, which varies in thickness according to the strength requirements of that particular portion of the tank. An extension on the lower end of the tank permits rod-drive access to the core through the 7.5-ft-thick subpile room ceiling.

The top cover plate is made of aluminum and is 2 in. thick. It is held in place by forty-two 1-in.-diam stainless steel bolts, which penetrate both the top cover plate and the top flange. The access cover plate, which closes the 24- by 29-in. access hole in the top cover plate, is made of 2-in.-thick 304 stainless steel and is held in place by a "submarine door" type clamp, which maintains a seal against the top cover plate. Eleven experiment access openings in the top cover plate allow installation of experimental facilities with out-of-vessel instrumentation and control. No experimental installations penetrate the access cover plate, so it retains its quick-opening, easy-access features.

The upper tank section is uniform in cross section from its upper part down to the top of the active lattice. The east quadrant of the tank, which consists of the region defined by 45° on both sides of east, has a wall thickness of $\frac{7}{8}$ in., and the rest of the tank has a wall thickness of $1\frac{3}{4}$ in. The thickness of the flattened side of the tank was increased to provide for extra strength in that section, because it is subjected to large stresses. Horizontal ribs $1\frac{3}{4}$ in. thick by $4\frac{1}{2}$ in. wide also add structural strength to the flattened side of the vessel. Coolant enters the reactor in this upper tank section by two diametrically opposed inlet water lines located near the top of the vessel on its north and south sides.

The middle section of the tank, which encloses the core region, is $\frac{7}{8}$ in. thick on the east half of the tank and $1\frac{3}{4}$ in. thick on the west or flattened half. The thickness of parts of the north and south walls of the tank is less in the region below the coolant inlet lines, around which the extra strength characteristics of the thicker walls were desirable.

There are nine large penetrations of the reactor vessel in the middle section. Six beam-hole facilities (Sect. 5.3.3) penetrate the vessel on its east face, and two large engineering facilities (Sect. 5.3.2) penetrate each of the north and south sides of the vessel. On the west side of the vessel and next to the core there is a 27-in.-wide by 30-in.-high by $5\frac{3}{8}$-in.-deep "window" that is inset into the reactor vessel (Fig. 1.1) and is used as an out-of-vessel high-flux experimental facility (Sect. 5.3.1). The walls and the face of the inset are $1\frac{3}{4}$-in.-thick aluminum.

The lower section of the tank is cylindrical, with an inside diameter of ~64 in. and a wall thickness of $\frac{3}{4}$ in. There is one large penetration of the tank in this lower section, and that is a 16-in.-ID inspection port located on the east side of the vessel. This port is closed with a blank flange, but it may be opened to provide access to the region immediately below the reactor core.

The lowest part of the vessel is an ~60-in.-ID aluminum cylinder, whose bottom forms the bottom plug of the reactor. The two coolant exit lines emerge from this section in diametrically opposite directions (north and south), and the control-rod drives penetrate the lower plug.

All of the reactor vessel penetrations are listed in Table 5.8.

Table 5.8. Reactor Vessel Penetrations

Description	Number	Size (in.)
Top head	1	See Sect. 5.3
Top access cover[a]	1	24×29
Top experiment access[a]	4	8 ID
Top experiment access[a]	4	5 ID
Top experiment access[a]	1	$4\frac{1}{2}$ ID
Top experiment access[a]	1	4 ID
Top experiment access (V-11)[a]	1	$3\frac{3}{8} \times 7\frac{1}{4}$
Slant access facilities	2	6 ID
Horizontal access holes	4	$2\frac{1}{2}$ ID
Horizontal access hole	1	2 ID
Horizontal access hole	2	$1\frac{1}{2}$ ID
Cooling water inlets	2	18 OD
Reflector access pipes	4	$\frac{1}{2}$ ID
Beam holes	6	6 ID
Engineering facilities	2	19×25
Poolside experimental facility	1	27×30
Inspection facility	1	16 ID
Poolside-facility coolant pipe openings	2	6 ID
Coolant exits	2	18 OD
Drive-rod penetrations[b]	6	2 ID
	48	

[a]Located in top head.

[b]Located in bottom plug.

5.3.1 Poolside Experimental Facility

In this, the most accessible facility in the reactor (Fig. 5.12), experiments may be placed on the flat west side of the reactor tank close to the lattice. The maximum thermal and fast (>0.4 Mev) neutron fluxes available at this facility are each $\sim 4 \times 10^{13}$ neutrons cm^{-2} sec^{-1}.

5.3.2 Engineering Facilities

These two large facilities (Fig. 5.12), located on the north and the south sides of the reactor, each have 19- by 25-in. obround access holes to the side of the lattice and are closed with $5\frac{1}{2}$-ft-diam shielding plugs. These plugs may be penetrated with several smaller holes for experiments that individually do not require the use of the entire hole. Such penetrations are designated HN-1, HN-2, etc. (for the north facility). The maximum thermal-neutron flux in these facilities is about 7×10^{13} neutrons cm^{-2} sec^{-1}.

5.3.3 Horizontal Beam Holes

Six beam holes, each 6 in. in inside diameter, are provided. The ORR beam holes are of circular cross section, decreasing in outside diameter from approximately $9\frac{1}{8}$ in. at the outer face of the concrete shield to approximately $6\frac{3}{4}$ in. at the reactor core box. Each beam hole is equipped with a collimator plug for operation as an experimental facility and, in its absence, a dummy plug. The plug seals the outer end of the beam hole and extends inward to within approximately 3 ft of the reactor vessel. Service piping enters through the plug.

During installation or operation of experiments at the beam holes, it sometimes becomes necessary to fill the liner, the plug, or both with water to provide biological shielding at the first level. The beam can be shut off by flooding the 6.4-ft-long

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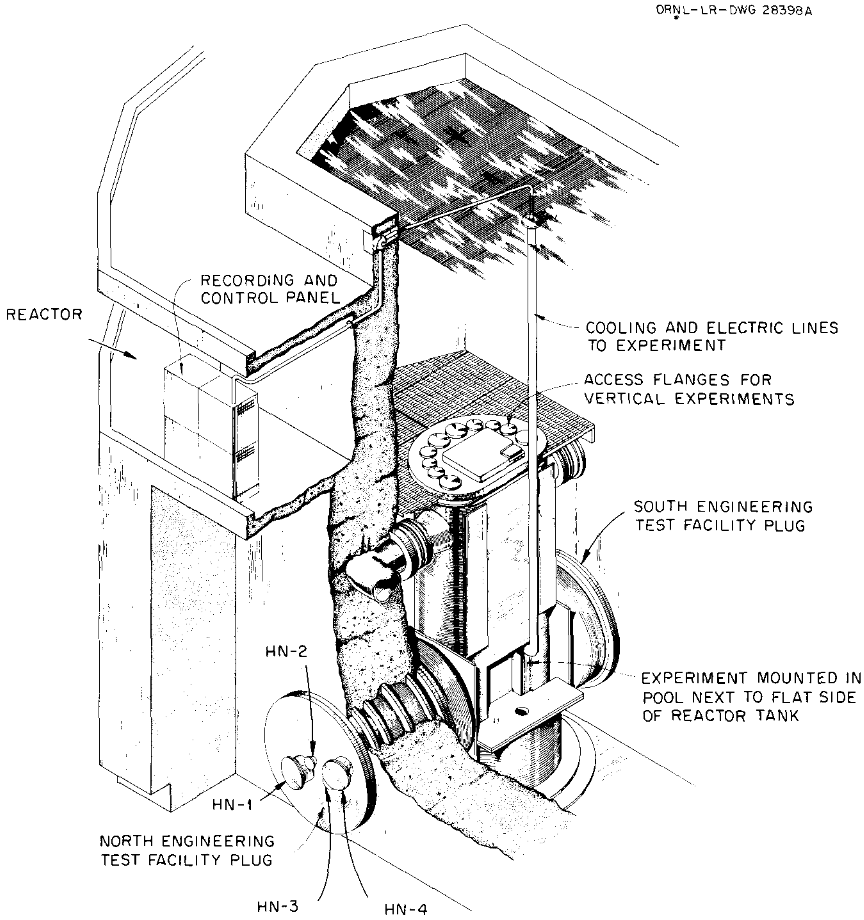

Fig. 5.12. Engineering-Test and Poolside Irradiation Facilities.

collimator section of the hole with water and lowering a lead-filled shutter. Subsequently, the plug and/or liner must be drained, dried, and possibly placed under continuous air or helium purge, depending on the desires of the experimenters.

Six service-piping headers are located in the basement adjacent to the east end of the reactor wall and directly under the beam holes. These headers provide

1. water supply from the pool cooling system,
2. water return to the pool cooling system,
3. plant air reduced to 15 psi,
4. warm drain,
5. off-gas,
6. helium supply from outside the building.

Lines from the above-listed headers lead to each liner and plug in such a way that they can each, separately, be

1. filled with water, air, or helium,
2. purged with air or helium (continuously),
3. filled with continuously recirculated water.

5.4 Pools

The reactor pools consist of a reinforced-concrete, aluminum-lined 10-ft wide tank, with a 39 ft long by 26 ft 6 in. deep storage section and a 21 ft 10 in. long by 28 ft 8 in. deep reactor section. The reactor section and storage section are separated by a removable two-piece watertight gate, and the storage section is divided into two equal-sized pools by another identical gate.

The length of the reactor pool section is sufficient for experiments to be conveniently inserted and removed on the side of the reactor opposite the beam-hole face. The storage pool length was determined by the remaining distance to the west wall of the building, allowance being made for a passageway around the end of the pool. The pool is used to store fuel elements and all irradiated items of equipment, and may serve as an experiment area involving high-intensity sources. The separation of the storage area into two independent sections permits emptying and cleaning of the pools without removing all the pool contents to another building for interim storage. The pool gates are of aluminum and stainless steel construction, are rubber-sealed, and are designed to withstand hydrostatic pressures equivalent to water up to the full height of the gate on either side.

The pool width of 10 ft was more or less arbitrarily established on the basis of achieving an optimum economic combination shield of water and barytes concrete in the wall around the reactor (see Sect. 9).

The three pool sections are lined on the sides and bottom with $\frac{1}{4}$-in.-thick aluminum plate. An economic study was made of the relative costs of an aluminum liner vs an Amercoat paint coating, and the aluminum liner proved to be more economical with regard to long-term maintenance, in addition to separating the leakage problem from the concrete structural shielding problem. The liner on the pool sides is welded to aluminum structural members embedded in the concrete on about 3 ft 6 in. centers. A 2- by 8-in. aluminum bar on each side of the pool is welded to these structural members at about the midpool depth. These support bars were installed at the time of initial construction and were attached directly to the liner structural members rather than the liner shell itself. The bars are designed to support a uniform live load of 3 tons per foot of pool length. In the reactor pool section two additional 2- by 8-in. bars are provided at an elevation 4 ft 6 in. above the reactor center line. These additional bars support a catwalk made of aluminum grating at a convenient height for working on the reactor tank and provide convenient support for experiments placed around the reactor.

On the north and south sides of the pools there are scuppers which consist simply of gutters along the sides of the pools whose tops are level with the surface of the water. Water from the surface of the pool flows into the scuppers with a skimming action and is conveyed to the pool cleanup system and back to the pools. This action has the effect of removing dust and floating particles and contaminants from the pools.

Details of the construction of the pool walls are presented in Sect. 9.

5.4.1 Personnel Bridge

The personnel bridge is used to support personnel performing routine tasks which require that they be positioned vertically over their work in the pools and also to move items in the pools that must remain underwater. It spans the pools from north to south and is a 6-ft-wide, 14-ft-long platform with a 40-in.-high personnel railing around all sides. The floor is painted sheet steel and has removable panels to provide access through the floor to the region under the bridge.

The bridge is mounted on wheels and is driven on a track in the east-west direction by a 2-hp motor and its associated reduction gear train. The bridge tracks run the entire length of the three pools, so the bridge may be positioned over any point in any of the three pools.

5.4.2 Storage Facilities

Storage racks, with 30 positions each, for irradiated fuel elements are provided in the reactor pool and the center pool, as shown in Fig. 5.13. A storage rack with eight positions for irradiated shim rods is provided in the center pool. Materials stored in these facilities are normally in the form of standard fuel units or shim rods; however, fissionable materials or experiments belonging to research groups may also be stored.

Fresh, unirradiated fuel is stored in the ORR vault, located in the basement. This vault contains an unshielded wooden rack and a 60-element metal rack. The metal rack consists of two 30-element sections. Storage of beryllium pieces, heavy water, or ordinary water in this area is strictly forbidden; approval must be obtained before other materials can be admitted.

The methods of handling fissionable materials of the reactor core must be approved by the ORNL Criticality Review Committee. The design of fuel-unit storage racks and transfer casks must be approved, in writing, by the chairman of that committee.

Storage of fissionable material belonging to research groups is more flexible. Special problems

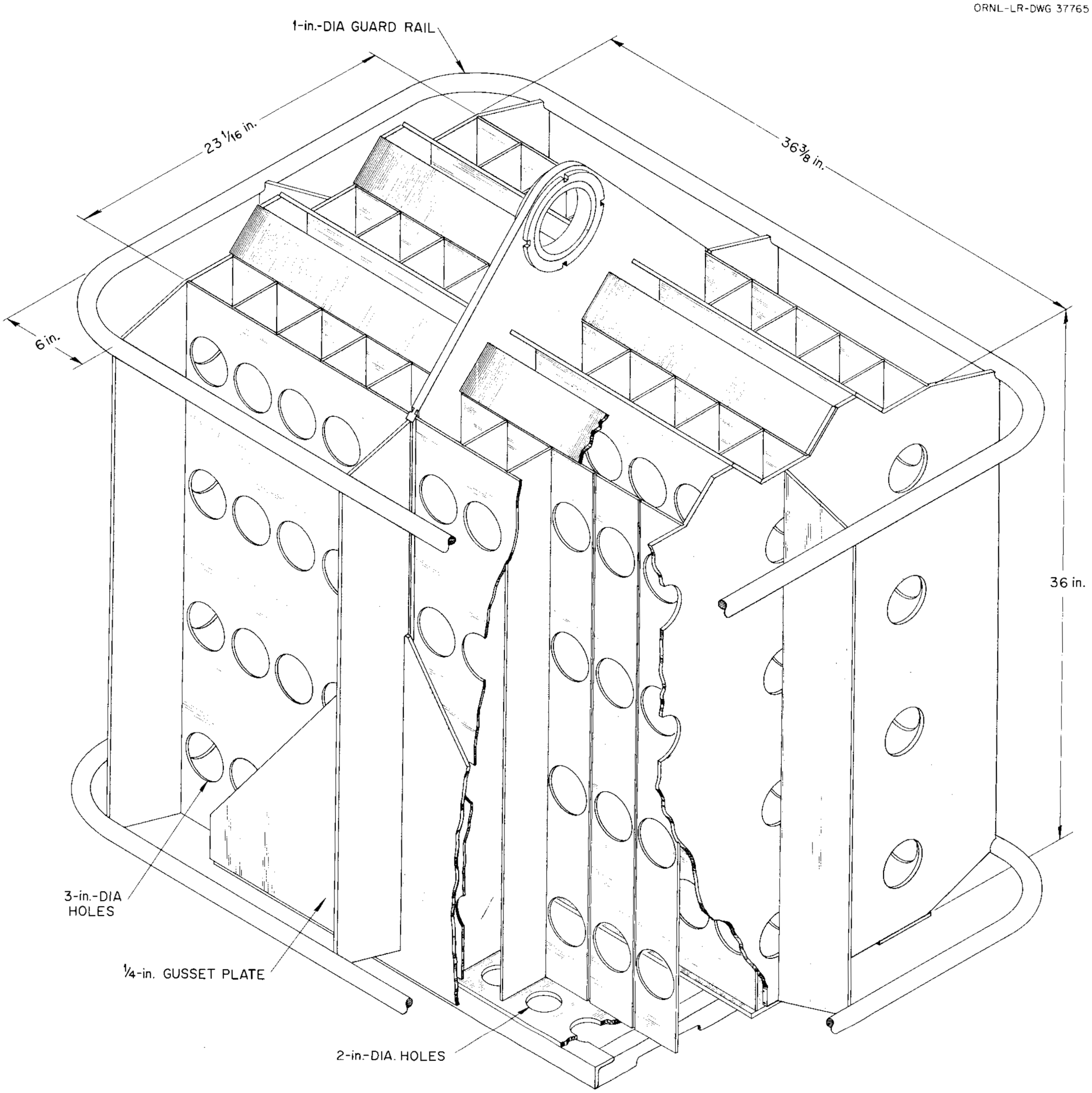

Fig. 5.13. Fuel Element Storage Rack.

associated with a particular experiment are brought to the attention of the ORR operating staff by the Technical Assistance Department, Operations Division. Special locations may be provided for storage of experiments, or the experiments may be suspended in the pool. Care is taken not to subvert criticality safety by suspending experiments containing fissionable material near standard storage racks.

5.4.3 Hot-Cell Facility

A hot cell, Fig. 5.14, is located above the west end of the storage pool and is arranged so that capsules or experimental rigs may be transferred from the pool into the cell through doors in its bottom. The cell is intended for use in disassembly and preliminary inspection of experimental rigs and capsules.

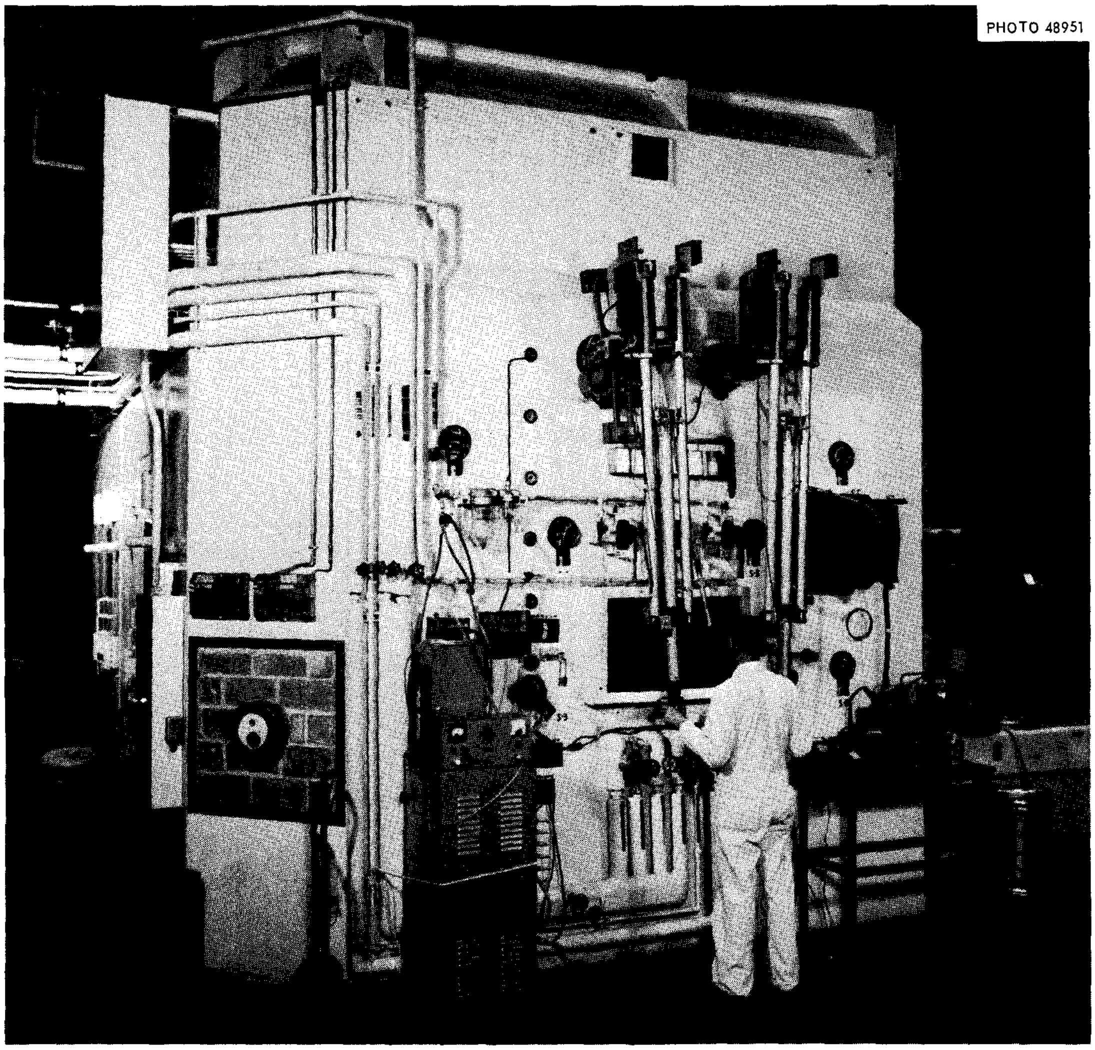

Fig. 5.14. Hot Cell.

The hot cell is divided into two sections, each of which has walls of dense concrete 3.5 ft thick, designed to shield 10^6 curies of ^{60}Co or the equivalent with a radiation level outside the cell of less than 5 mr/hr. The sections may be used separately only, but no shielding is provided between the two sections. The sections are completely lined with stainless steel, and only this stainless steel sheet separates the sections.

The cell is equipped with sprays for cleaning, hot drains, and hot off-gas connections. For experimental work, a pneumatic hoist, manipulators, and lead-glass windows are provided for the necessary handling and observation of irradiated materials inside the cell.

6. COOLING SYSTEM

6.1 Introduction

The cooling requirements for power operation of the ORR are provided by two separate cooling systems. One of these, the reactor cooling system, is designed to remove all of the approximately 30 Mw of energy produced in the core. The other, the pool cooling system, disposes of the small amount of heat transferred from the reactor tank to the reactor pool and the heat evolved from materials stored in the pools.

In the reactor cooling system, demineralized water as the primary coolant is pumped through the reactor tank at a flow rate of about 18,000 gpm. It passes through the shell side of four primary heat exchangers; at this point it transfers its heat to the secondary coolant, which is circulated through the tube side of the heat exchangers. The secondary coolant, treated process water, is then circulated through a conventional induced-draft cooling tower, which dissipates the heat to the atmosphere.

Approximately 0.6 Mw of reactor heat is transferred to the reactor pool by conduction from heated surfaces and by absorption of radiation. To accommodate this heat and up to 0.1 Mw of heat released by stored used fuel elements, a second cooling system, the pool cooling system, permits circulation of 700 gpm of pool water through the tube side of the pool heat exchanger. Here the secondary coolant, treated process water, absorbs the heat while circulating through the shell side of the heat exchanger. This secondary coolant is then circulated through a separate conventional induced-draft cooling tower and dissipates this heat to the atmosphere.

Because of the heat generated by the fission product inventory in the core, it is necessary to provide cooling to the core for a short time following shutdown. Under normal circumstances, this is handled by the primary circulation pumps. In the event of a failure of power to the main pump motors, adequate coolant flow is maintained by battery-powered dc motors attached to the shafts of the main coolant pumps. Any one of these three motors can provide coolant flow sufficient to prevent damage due to afterheat.

In association with each of the water systems are various components for demineralizing, degasifying, and flow measuring. Most of these components are located in the west end of the basement of the reactor building.

6.2 Reactor Cooling System

6.2.1 Primary System

As has been stated previously, the reactor primary cooling system has demineralized light water as the coolant. As illustrated in Figs. 6.1–6.3, the primary coolant enters the reactor vessel through two 18-in. lines near the top. The water flows down through the core, cooling it. This downward motion of the water helps to seat the fuel elements and does not oppose the scramming mechanism.

Water flows out of the bottom of the vessel through two 18-in. lines. These run into the concrete shielding, turn west until they are beyond the large experimental facility holes, and then turn upward. They continue upward in the concrete shield until they are above the level of the reactor core; then they make a 180° turn toward the basement. A line leading to a ball-float trap is tied into each of the 18-in. lines at the top of the 180° bend to disrupt any siphoning action when the water level in the tank drops to the level of the bend. This level is above the reactor core; consequently, the tank cannot be siphoned to a level below the top of the core.

The two 18-in. exit lines continue to the pipe chase in the basement, where they join into a common 24-in. line. This line runs out of the basement, increases to a 36-in. line, and extends in a northeasterly direction for about 200 ft. It changes size once more, back to a 24-in. line, before entering a 10,000-gal decay tank. The purpose of the tank, as well as the 36-in. line, is to delay transport of the water to allow any ^{16}N that may be present to decay to a safe radiation level.

After leaving the decay tank, the water proceeds through a 24-in. line to the primary system pump house. Beneath the pump house, the 24-in. line becomes a manifold which feeds three 16-in. lines and two 8-in. lines. Each of the 16-in. lines leads to the inlet of a 6000-gpm pump. These pumps provide the power to circulate the cooling water through the system. Battery-operated dc motors directly coupled to the 6000-gpm pumps continue rotation of the pumps during a power outage, so they provide afterheat cooling.

The two 8-in. lines leaving the manifold are for emergency cooling. One of these goes to a 1000-gpm electrically driven pump, which is used during reactor shutdowns. The other 8-in. line goes to a 1000-gpm gasoline-motor-driven pump, which provides additional afterheat cooling in support of the dc motors.

The discharge from the main pumps travels through a 24-in. line under the north wall of the pump house, where it branches into two lines. One line continues north to the eight water-to-air heat exchangers used for the former 20-Mw operation. This portion of the system is now valved off and is not in use.

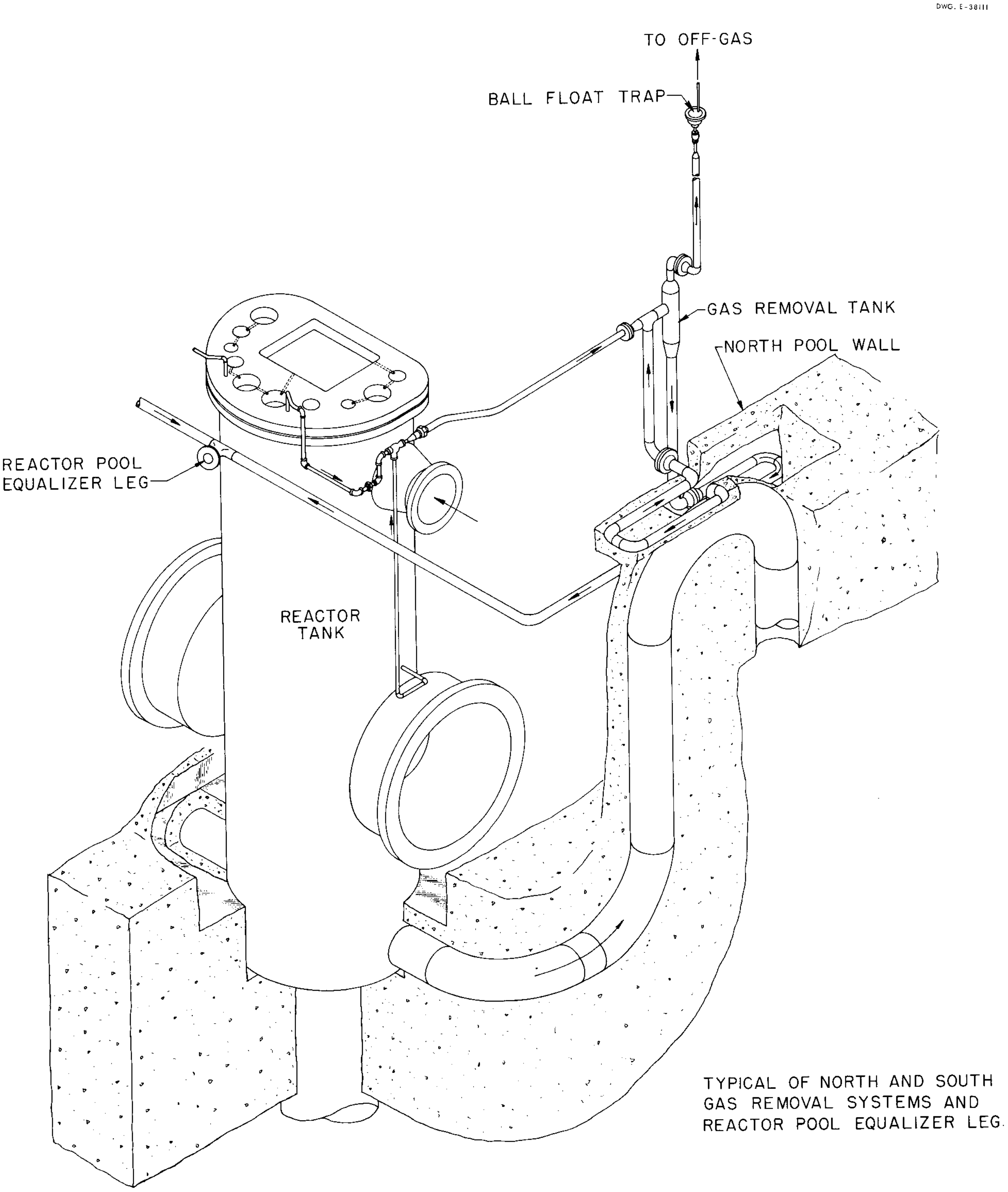

Fig. 6.1. Reactor Tank.

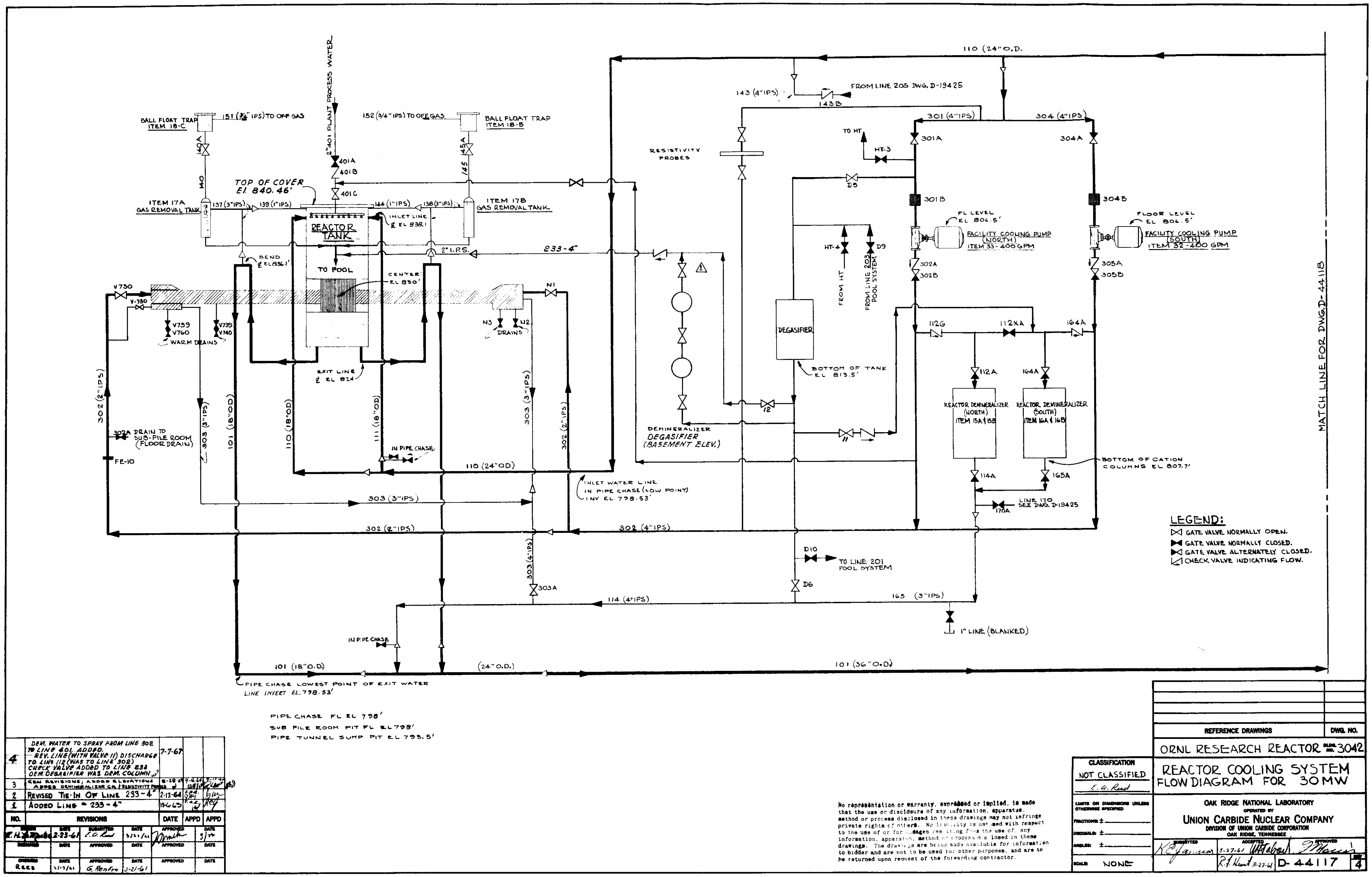

Fig. 6.2. Schematic Diagram of Reactor Primary and Secondary Cooling Systems.

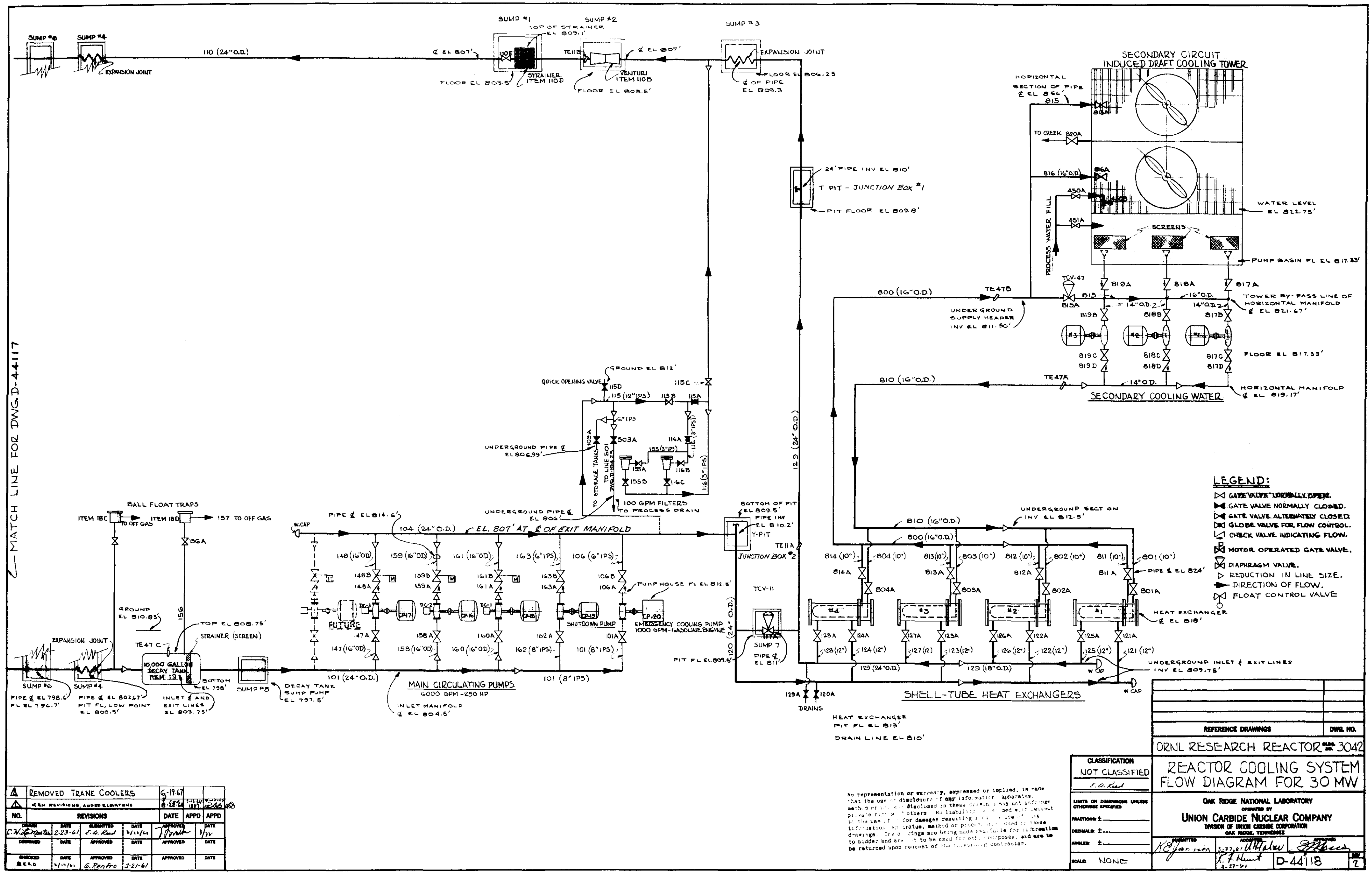

Fig. 6.3. Schematic Diagram of Reactor Primary and Secondary Cooling Systems, Continued.

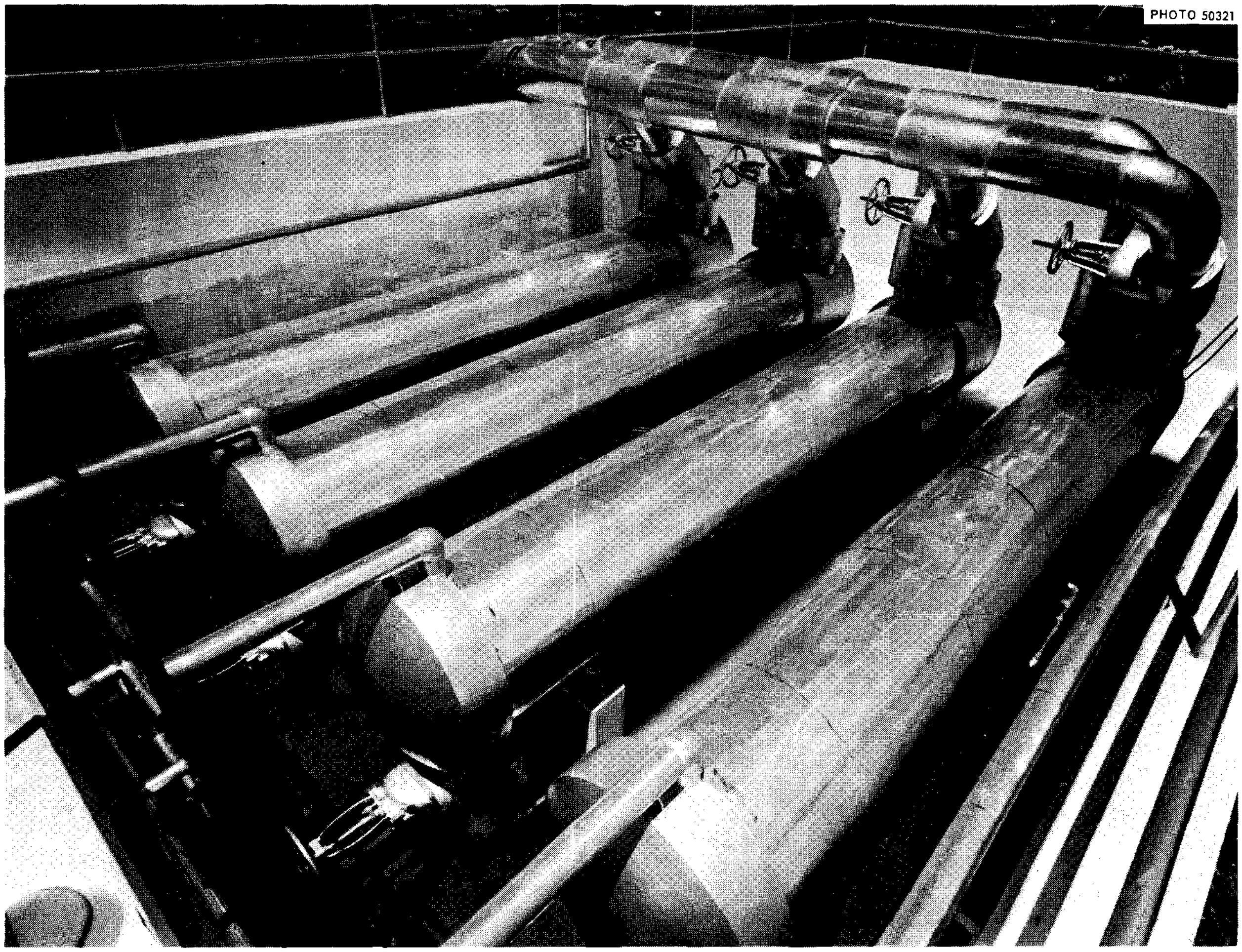

Fig. 6.4. Reactor Heat Exchangers.

The other line extends northeast to the heat-exchanger pit (Fig. 6.4). Here the heat generated in the core is transferred from the primary system to the secondary system by means of four stainless steel shell-and-tube heat exchangers connected in parallel. The reactor cooling water passes from east to west through the shell side of the exchangers.

A 24-in. wafer-type butterfly control valve with an air-motor positioner is installed in parallel with the four heat exchangers. This valve allows a portion of the cooling water to bypass the heat exchangers, depending on the cooling demand.

After leaving the heat-exchanger–bypass-valve arrangement, the water flows through a single 24-in. line, which continues back toward the pump house and converges with the discharge line from the water-to-air heat exchangers. The line then turns south under the porch of the pump house. A venturi in this section allows the rate of coolant flow to be monitored in the control room. The line then proceeds southwest, through the strainer, and back to the reactor building, where it enters the pipe chase and branches into two 18-in. lines that carry the cooling water through the concrete shield and up to the reactor inlets.

It is desirable to pass a portion of the reactor system water through a filter continually. Two 100-gpm filters, located north of the ORR pump house, have been provided for this purpose. These filters remove micron-sized particles.

6.2.2 Secondary System

The reactor secondary system uses process water as the heat-transfer medium. The secondary cooling water flows from the cooling tower basin, which normally contains approximately 70,000 gal of cool water, to the secondary system circulating pumps. The discharge from the pumps runs to a common 18-in. line in the secondary system pump house and proceeds westward to the outside. From this point, the line runs underground, southwest to the heat-exchanger pit, rises out of the ground, and enters the north top opening of the east end of the four shell-and-tube heat exchangers. The water makes two passes on the tube side and exits through the south top opening of the east end of the exchangers. The exit pipe parallels the inlet line to the heat exchangers and proceeds toward the secondary system pump house. Just outside the pump house, the line turns north, parallel to the cooling tower.

Part of the water can bypass the tower and proceed to the pump suction, the amount depending on the cooling demand; the remainder flows through the two risers to the top of the tower. As the hot water falls through the cooling tower, the cooling-tower fans draw air past it, evaporating some of the water and cooling the remainder. The cool water is then retained in the cooling-tower basin, which serves as a reservoir for the system. The basin water level is kept constant by a level-control valve, which adds process water to make up evaporation and other losses.

6.2.3 Cooling Control System

The objectives of the temperature-control system are as follows:

1. To maintain steady-state control of the reactor inlet temperature.
   a) It is important to minimize the fatiguing of the reactor structure which results from stresses produced by temperature cycling.
   b) The most accurate method of obtaining reactor power is to compute it from reactor cooling water flow and the temperature rise across the reactor. There is a transport delay across the reactor; consequently, if the inlet temperature is changing, the measurement of the temperature differential, and therefore the calculated power, is incorrect. The temperature differential is only 11 to 12°F at the 30-Mw power level; an error in the differential temperature measurement of 0.5°F will result in a 4% error in calculated power.
2. To provide a system of sufficiently fast response so that the reactor operation will not be inhibited by the temperature control system.
   a) When the reactor is unexpectedly shut down by abnormal conditions, xenon poisoning makes it mandatory to return to power within a short time or else suffer a delay for a change of reactor fuel. The cooling system must, therefore, be capable of control with a reactor power cycle from the 30-Mw power level to essentially zero power and, thence, rapidly back to the 30-Mw power level.
   b) Good transient response reduces the danger of overheating the reactor fuel and reactor structure on rapid power increases. The operating conditions at 30 Mw are such that very little temperature overshoot is permissible.
   c) A fast response system is required to prevent recirculation of high-temperature slugs of water around the primary cooling loop. The long loop time and low natural damping enhance the problem, as has been exhibited in the past.
3. To provide a design such that the failure of control valves within the water loops will not result in an abrupt loss of cooling to the reactor. This dictates the use of "fail-closed" bypass valves rather than a series of throttling valves in the cooling loops. Thus, the most probable valve failure will not stop cooling water flow and will allow maximum cooling.

Three systems are used:

1. Tower control system – controlling the temperature of the cooling-tower basin.
2. Secondary control system – controlling the mean temperature of the secondary side of the heat exchanger.
3. Primary control system – controlling the temperature of the water returning to the reactor.

The ultimate objective is ±0.25°F control of the primary water passing through the reactor. The three systems, in the order listed, lead progressively toward this end; that is, control of the tower-basin temperature allows the secondary temperature to be regulated, and control of the secondary allows close control of the primary temperature. In this way, fast response and a fine degree of control are attained.

All three of these control systems are designed around simple temperature control. The primary control system has a fixed set point designed to maintain a constant cooling water temperature determined by the desired power level of the reactor. The constant temperature is accomplished by adjusting, by means of a bypass valve, the amount of cooling water flowing through the primary-to-secondary heat exchanger. At the other end of the cooling system, the tower basin is also designed to operate at a constant temperature. The basin may be regarded as an infinite heat sink from which the primary system may draw varying amounts of cooling capacity as the situation demands. Its temperature is maintained slightly below the level required for dissipation of full reactor power, and thus cooling capacity is immediately available. In between is the primary-intermediate link. It provides for transference of the cooling potential from the basin to the primary loop. In effecting the transfer, the temperature set point of the secondary is made to vary in response to conditions from two sources: (1) fluctuations in the reactor power level – providing more cooling water from the basin if the power goes above the desired level and less if it goes below, and (2) adjustments already made in the primary system.

When a fluctuation occurs in the power level, the primary control system responds immediately, opening or closing the secondary bypass valve to meet the new demand. The secondary control system then continues to alter its set point by changing fan speeds until no correction is necessary by the primary control system. This allows the secondary bypass valve to return to its equilibrium position, again ready for optimum response in either direction to sudden fluctuations in power level. Summation of the two conditions determining the secondary control set point is made by a simple pneumatic analog computer. The computer output is used to regulate the secondary bypass valve.

Thus by the interconnection of three control systems, the conditions for rapid and optimum control response are provided in the primary control system. The system adapts the slow response of the tower basin to the fast responses of the reactor and is able to quickly damp out transients.

6.3 Pool Cooling System

6.3.1 Primary System

The heat that is transferred to the pool surrounding the reactor tank is approximately 0.7 Mw, as discussed in Sect. 6.1. The purpose of the pool primary system (Fig. 6.5) is to transfer this heat by way of a shell-and-tube heat exchanger to a secondary system, where the heat is dissipated through an induced-draft cooling tower.

The water leaves the pool through a 6-in. line and travels from the reactor pool to the pool-coolant pump, located in the west end of the basement (Fig. 6.6). The pump, a 900-gpm unit, forces the water through a filter arrangement to the tube side of the shell-and-tube heat exchanger, located near the pump. The water then returns to the reactor pool by three separate paths, as described below.

During the design of the ORR, there was considerable concern about the formation of the radioisotope ^{16}N in the reactor pool adjacent to the reactor. Radiation readings of ~ 20 r have been obtained from ^{16}N-bearing water in the exit water line from the reactor core. Nitrogen-16 has a half-life of 7 sec, but it was feared that the natural convection around the reactor would move the hot gas to the pool surface before it could decay sufficiently to be harmless. Sufficient delay was accomplished by a system of jets which directed approximately 140 gpm of pool return cooling water downward along the outside of the reactor tank wall, as shown in Fig. 6.7.

Another source of relatively high ^{16}N activity was at the reactor pool side. The delay here was accomplished by installing a jet eductor, Fig. 6.8, and using 140 gpm of the pool return cooling water to provide a suction at the pool side. The ^{16}N-bearing water is pulled down by the suction and dispersed into the reactor pool. With both systems in operation, the radiation level is only about 8 mr/hr at the pool surface.

Since the beam-hole liners are anchored both at the reactor tank and at the concrete shield, they must be water-cooled to prevent excessive stresses from developing in the reactor vessel due to temperature variations. Constant beam-hole liner temperatures are assured by spraying water from manifolds on the top and bottom of each liner. Water for these sprays is furnished by taking part of the water returning from the pool

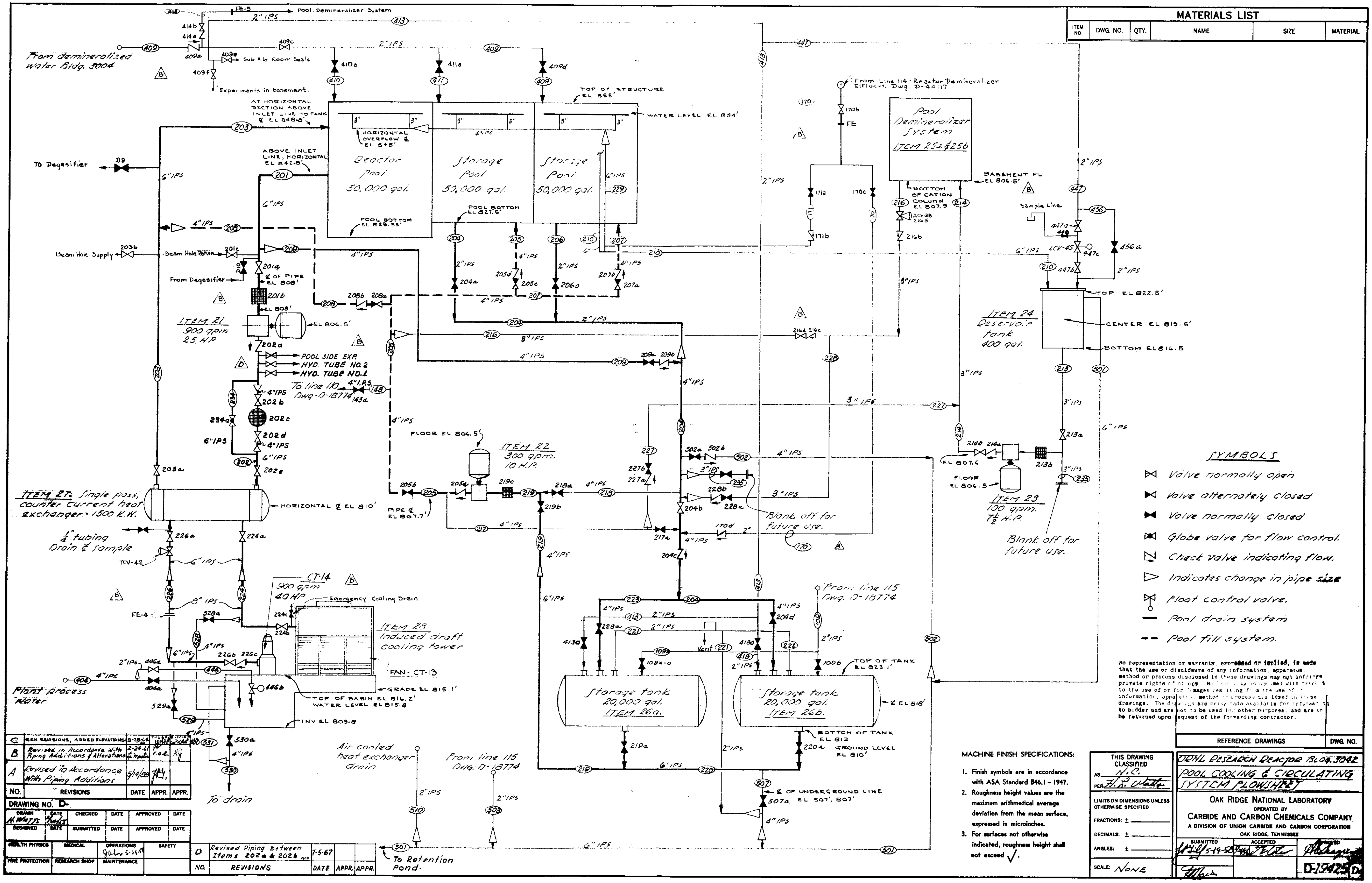

Fig. 6.5. Reactor Pool Cooling System.

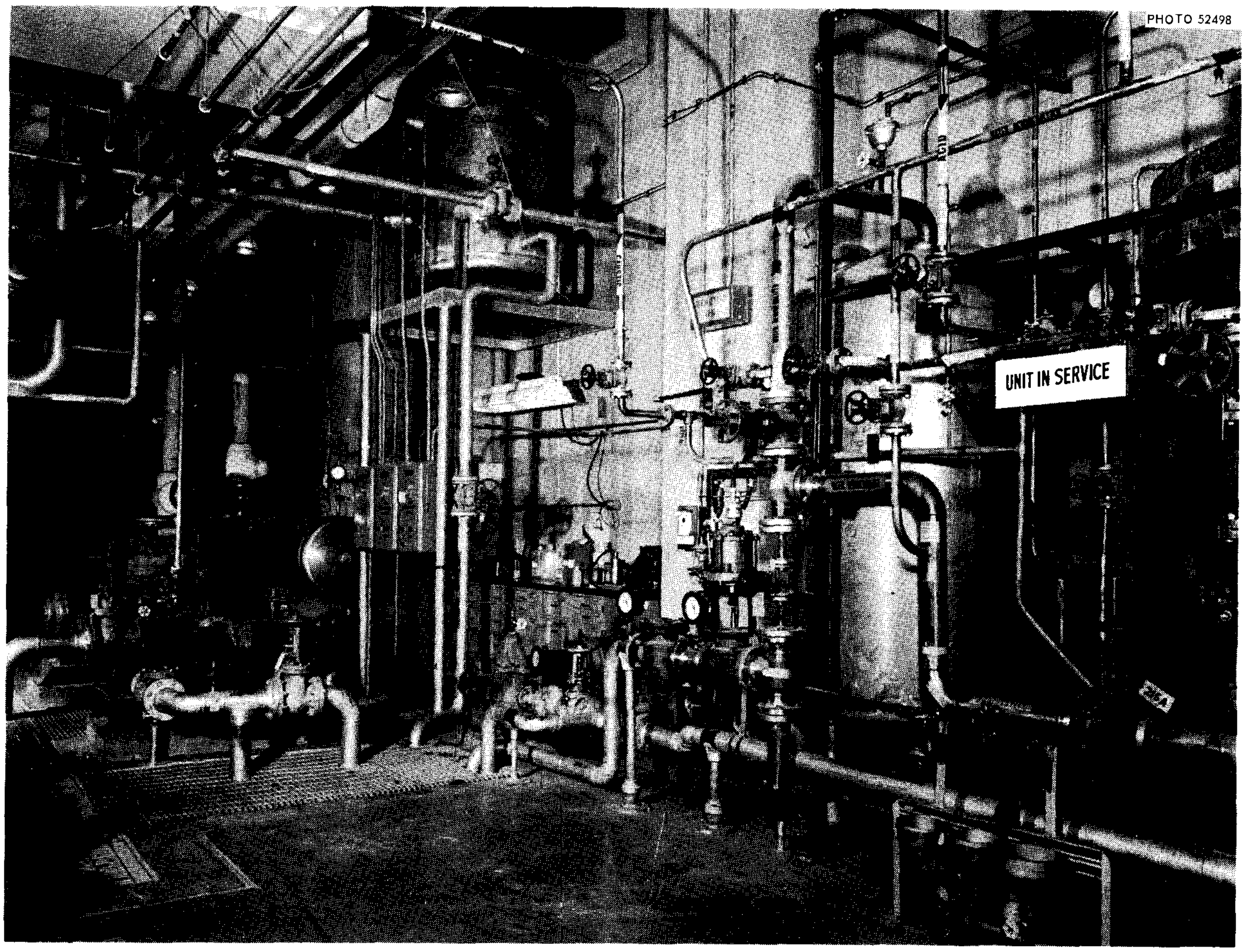

Fig. 6.6. Pool Heat Exchanger and Pump.

cooling heat exchanger and directing it to the spray manifolds, as shown in Fig. 6.9. About two-thirds of the total pool cooling flow is sprayed on the beam-hole liners. The temperature of this returning water is about 85°F.

The reactor cannot be operated at full power unless the beam-hole liner spray cooling system is in operation. A relay in the pool cooling water flowmeter prevents raising the reactor above 1.8% of full power without adequate cooling flow. Since the beam-hole liner spray system is the only pool cooling water return line that contains no cutoff valve, operation of the liner spray at any significant reactor power is ensured.

6.3.2 Secondary System

The heat from the pool primary system is transferred to the treated process water in the pool heat exchanger, as previously mentioned. The process water from the heat exchanger flows through a 6-in. line to a cooling tower located west of the primary pump house. The water is pumped out of the tower basin and back to the shell side of the heat exchanger. The volume of water in the piping of the loop is 2410 gal, with the cooling-tower basin and sump bringing the total capacity to 5270 gal.

6.3.3 Cooling Control System

The temperature of the demineralized water returning to the reactor pool from the heat exchanger is controlled by an automatic system located on the panel adjacent to the heat exchanger in the basement. This temperature is recorded at the basement panel and indicated on a gage in the control room. The normal control point is 88°F,

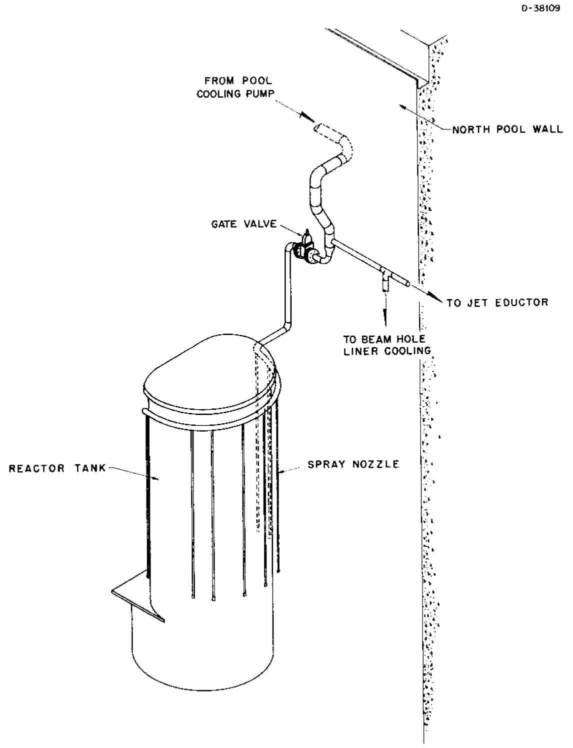

Fig. 6.7. ^{16}N Dispersion System.

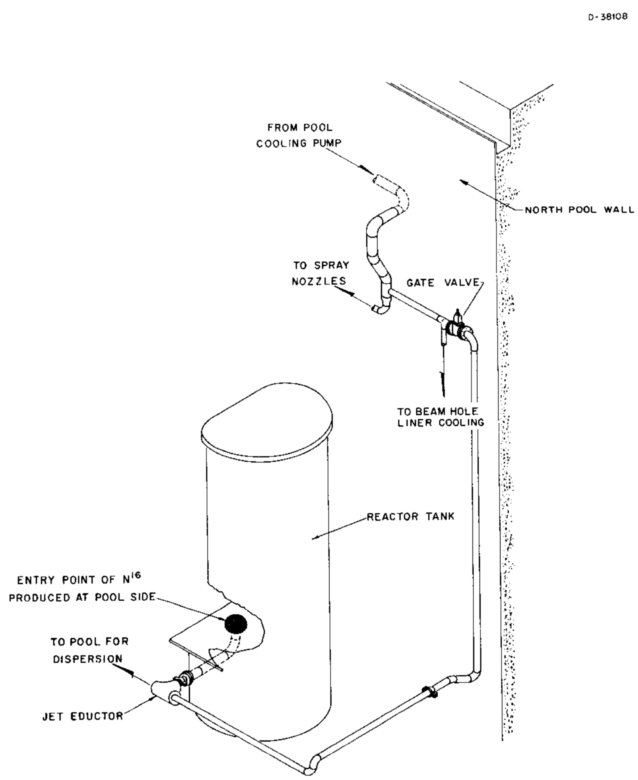

Fig. 6.8. ^{16}N Dispersion System, Pool Side.

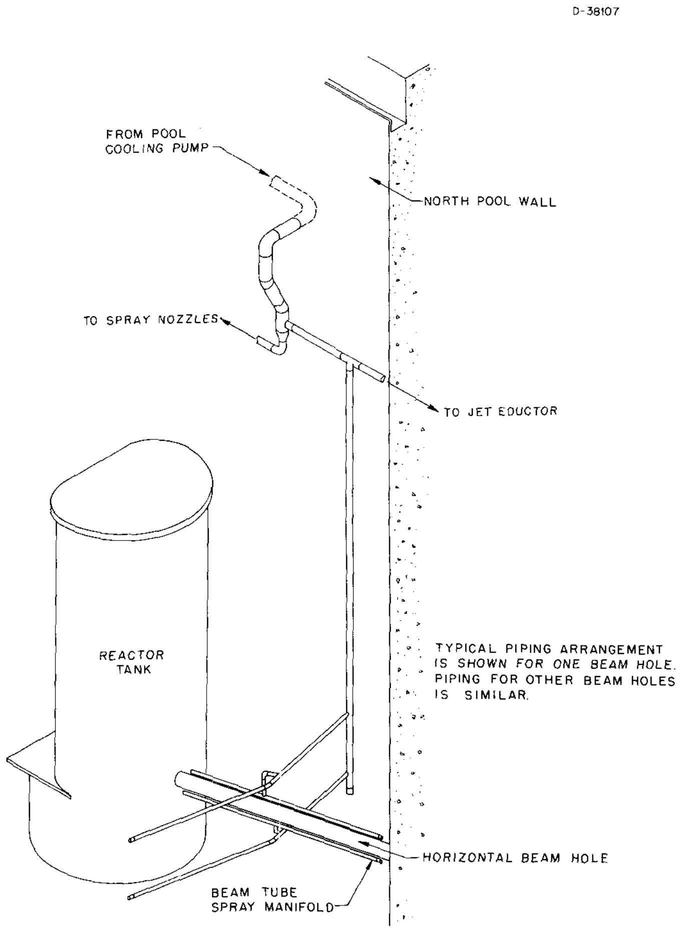

Fig. 6.9. Spray Cooling for Beam-Hole Liners.

with the pool high-temperature annunciator activated at 93°F.

The pneumatic throttling valve in the secondary cooling loop may be positioned automatically or manually. The mode of valve control is selected by a transfer lever on the basement control panel.

In the automatic fan controller, cam switches are operated by a motor, which is turned on or off by pressure switches connected to the same pneumatic line as the throttling valve. As the air pressure to the temperature control valve is decreased to 4 psi, the valve is allowing maximum cooling; therefore, fan speed must be increased. A switch closes at 4 psi to drive the cam-switch assembly and thereby increase fan speed. The fan starts on low at 60° rotation and shifts to high at 120° rotation. Motor-limit switches are located at 1 and 120°.

If the cooling is excessive, air pressure to the temperature-control valve will increase and close the valve. At 9 psi a switch will close to drive the cam-switch assembly and reduce fan speed. However, as long as the pneumatic pressure to the temperature-control valve is between 4 and 9 psi, the fan speed will not be changed; the valve will provide the necessary control.

6.4 Auxiliary Water Systems

6.4.1 Demineralizers

In reactor cooling systems, the radioactivity level, the rate of corrosion, and the rate of deposit formation must be closely controlled. At the ORR, these objectives are met by demineralizing the cooling water with multiple-bed ion exchange resins. There are three such systems in operation as follows.

Reactor Demineralizers. – There are two demineralizer systems: one on stream and the other kept on standby, regenerated and ready for service. Both systems can be used simultaneously if conditions warrant it, for example, after a long shutdown when the water has been subjected to more impurities than during normal operation.

Most of the components in the reactor cooling loop are made of aluminum, although some of the smaller items are stainless steel. The portion of these materials which dissolves in the water, dissolved gases, and fission products which diffuse from the fuel cladding constitute most of the impurities found in the primary loop. A slight amount of demineralized water is added to the system continuously during normal operation to replace that lost through cooling and sealing various bearings and glands. It is essential that the primary water have as few impurities as possible, since it passes through the high-neutron-flux core region, where impurities become highly activated; so some cooling water is continually bypassed through the demineralizers to remove trace impurities.

Three checks are available for judging the purity of the water in the reactor system. The first is the radioactivity level measured in disintegrations per minute per milliliter. The second is pH, a measure of the acidic or basic quality of the water. The third is the electrical resistance or resistivity of the water. Measures of pH and resistance are made once each shift; water activity is measured once each day. Resistivities and activities are also recorded continuously in the control room. In general practice, the water resistivity and the water activity checks have been the best guides to the effectiveness of a demineralizer unit and the need for regeneration.

The reactor demineralizers are located in the basement at the west end of the building. The two systems are separated from each other and enclosed behind 6-in. solid-concrete-block walls. In addition to the concrete shielding, the cation columns are shielded with approximately 4 in. of lead. Extension handles on all valves extend through the block walls, so that remote operation is possible even though it is not normally required by the radiation level.

Each demineralizer system consists of two resin columns: a cation column, 30 in. in diameter with 72-in. straight sides, containing 10 ft^3 of resin, and an anion column, 40 in. in diameter with 96-in. straight sides, containing 37 ft^3 of resin. All resins are standard commercial types, namely, Rohm and Haas Company Amberlite IR-120 cation resin and Amberlite IRA-401 anion resin.

The driving force causing the water to pass through the demineralizers comes from the facility coolant pumps. The system is so piped that either pump, north or south, can be used to feed either demineralizer by use of the special crossover valve. However, it is preferred that this valve remain closed and the flow be directed from the north pump to the north demineralizer or the south pump to the south demineralizer.

Pool Demineralizer. – The radioactivity in the ORR pool water would build up to an appreciable level if a method were not provided to remove the radioactive ions. The neutron flux in the pool just outside the reactor tank wall is of the order of 10^{13} neutrons cm^{-2} sec^{-1} and produces ^{24}Na, ^{16}N, and other radioactive nuclides; so a pool bypass demineralizer is used to remove a portion of these nuclides.

In the present operation of the demineralizer, water overflow from the pool is collected in a 400-gal overflow reservoir, which serves as the supply for the pump in the demineralizer system. The pump passes the water through both a cation resin bed and an anion resin bed, then returns it to the pool. The cation unit, 48 in. in diameter and 72 in. high, contains 40 ft^3 of IR-120 cation resin. The anion unit, also 48 in. in diameter and 72 in. high, contains 35 ft^3 of IRA-401 anion resin. The system, using both the cation and anion columns at all times, has a flow rate of 100 gpm, with the discharge of the system reentering the pool through the fill lines.

Degasifier Demineralizer. – A third demineralizer unit for the reactor primary water system, consisting of an anion and a cation column, is used to purify the effluent from the degasifier (see Sect. 6.4.2). The old north mixed-bed demineralizer (originally in series with the reactor's north demineralizer units) is now the cation column for the new demineralizer; the south mixed-bed demineralizer (originally in series with the reactor's south demineralizer units) is now the anion column for the degasifier demineralizer.

The water supply for the new demineralizer is taken from the water discharge line of the degasifier unit serving the reactor primary water system. The water demineralized by the new unit is routed to the pressure-equalizer line of the reactor tank. The primary function of this system is to decontaminate any water that expands from the reactor water system into the pool. An additional advantage is extra demineralizer capacity for the reactor water system. Prior to the installation of the new demineralizer, the expanding water was discharged directly into the pool. It should be noted that the degasifier demineralizer cannot be placed in service unless the degasifier is in service. When the demineralizer is removed from service to regenerate the resin columns, the degasifier effluent is routed directly to the reactor cooling system, thereby permitting a continuation of the degassing of the reactor water.

6.4.2 Degasifier

The ORR bypass degasifier is a unit designed to remove entrapped gases from the water used in the reactor and pool primary cooling systems. The use of this unit aids in reducing the air radioactivity in the reactor building. The degasifier unit is located in the basement, east of the north reactor anion column enclosure, and is housed in a cubicle with 8 in. of solid-concrete-block shielding.

A stream of water is bled from either the reactor system, the pool system, or the hydraulic-tube system into the degasifier system through an automatic control valve. The stream is sprayed into the top of the degasifier tank and collects as a condensate at the bottom, where it enters a centrifugal pump which returns the water to the reactor system through the exit line from the reactor demineralizers, to the reactor system via the line to the equalizer leg, to the pool cooling system via the suction of the pool cooling pumps, or to the demineralizers via the inlet line to the demineralizers (see Fig. 6.10). The function of the automatic control valve mentioned above is to maintain the water in the tank at a level which will prevent cavitation of the degasifier pump.

To remove any liberated gases, the degasifier tank is evacuated continuously by a steam jet, which produces a negative pressure of 26 in. Hg. Located between the jet and the degasifier tank is a condenser, which removes water vapor from the evolved gases. This condensed vapor is returned to the degasifier tank via the same line supplying vapor to the condenser, since the pipeline is large enough in diameter to allow simultaneous flow in both directions.

The exhaust from the steam-jet vacuum pump, which now contains the gases removed from the water, is fed through another condenser to condense the steam. The condensate and the free gases enter a fluid-gas separator via separate lines. The final separation of liquid and gas is made here, with the gas being removed by a line connected to the building off-gas system. The fluid, since it might contain traces of radioactivity, is piped to the "hot" drain in the pipe chase. The condensers are cooled by plant process water.

6.4.3 Facility Cooling System

Reactor-system cooling water, supplied via the facility cooling pumps, is used to cool the north and south experimental facilities. The pebble bed of the large plug, the annulus of the plug, and the dished head and oval facility in the reactor vessel are normally filled with this water, which circulates through the components and removes heat.

In view of the fact that there are similar units on opposite sides of the reactor vessel, the vessel can be subjected to unequal hydraulic pressures on the two sides to cause an undesirable net force in one direction; therefore, extreme caution must be exercised when changing the water flows to these facilities. Design engineers have placed a maximum differential pressure on the facilities of 15 psi, and this value should never be exceeded. The piping to the facilities has been arranged to simplify draining and to minimize the chance of error.

It is recognized that serious radiation incidents can occur as a result of errors during draining, since the reactor pool water is usually at grating or tank-top level when the facilities are drained. It is possible to lower the water level in the reactor tank during an erroneous draining procedure, resulting in an extremely high background in the reactor pool. In order to prevent misoperation of the facility cooling system, administrative requirements and procedures have been established for filling and draining the experimental facilities.

6.4.4 Gas Removal System

The ORR gas removal system consists of several ball-float traps which are located above, and connected to, parts of the reactor water system

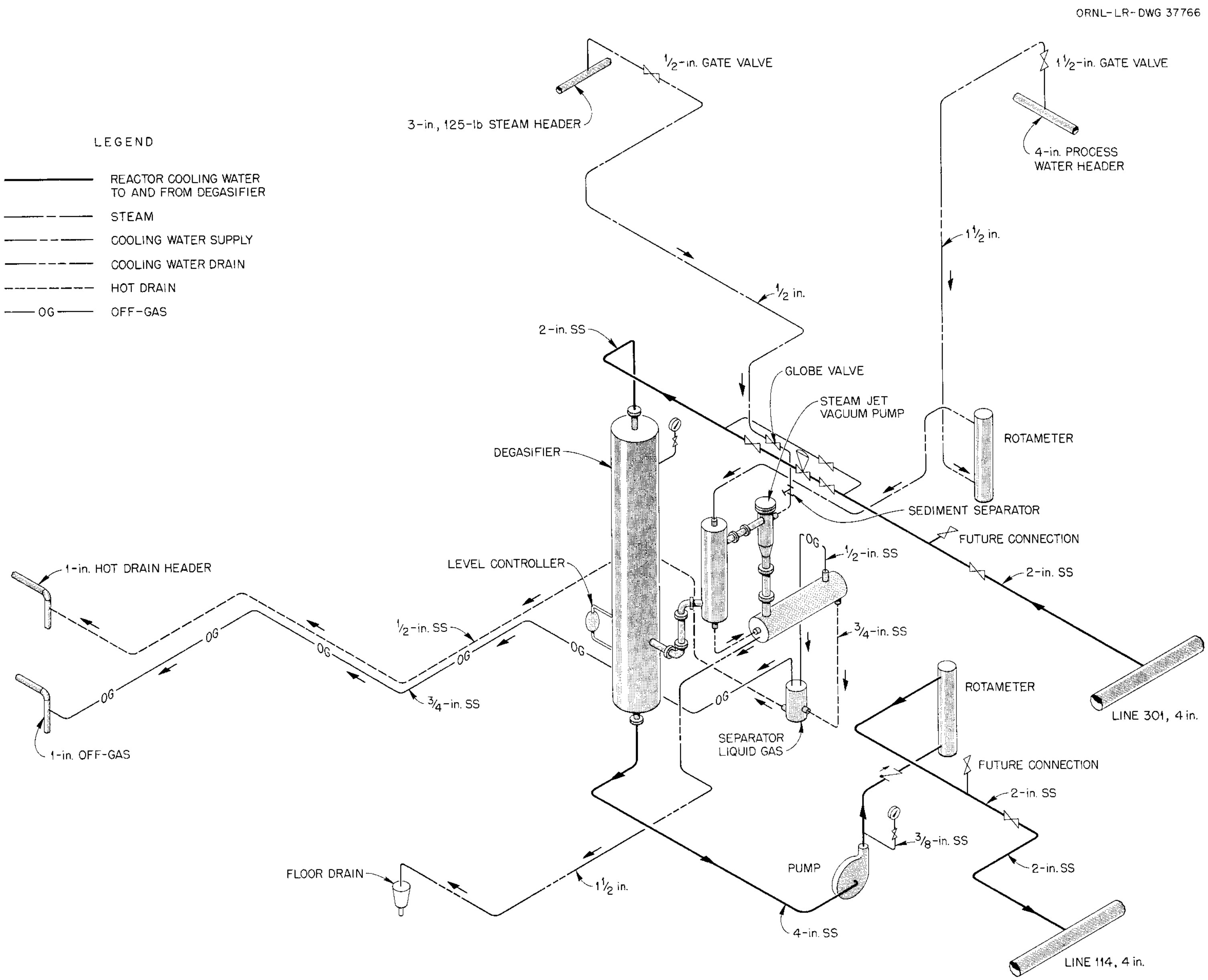

Fig. 6.10. Degasifier Flow Diagram.

where gases would naturally collect. These ball-float traps, normally full of water, contain valves which open when gas is collected. The exit lines are connected to the building off-gas system.

Figure 6.1, the gas removal system of the ORR, shows the ball-float trap and associated piping for gas removal from the reactor tank and the north exit water line. A similar installation exists for the south exit line.

The ball-float trap is located in a recess in the pool parapet. The gas removal tank is located in the reactor pool below the grating level and near the north wall of the pool. Any entrapped gases would move, as shown by the arrows, from the reactor tank or the exit line to the gas removal tank, up to the ball-float trap, and from there to the off-gas system.

The reactor pool equalizer leg, also shown in Fig. 6.1, is not directly related to gas removal but rather allows for expansion or contraction of water in the otherwise totally enclosed reactor cooling loop. The equalizer leg can also be used to provide emergency cooling flow.

Additional ball-float trap installations throughout the reactor water system are:

1. Above the bellows-type expansion joint in line 101, reactor water exit. This unit is located in the expansion pit (sump No. 5) near the bottom of the steps to the pump house area.
2. Above the 10,000-gal decay tank located south of the pump house.
3. One above each section of the reactor demineralizers (total of 6) and one above each section of the pool demineralizers (total of 2).

6.4.5 Demineralized Water Makeup

A continuous supply of fresh demineralized water is necessary to supply the shim-rod-drive seals and the bottom-plug seal in the subpile room, service those reactor experiments for which an uninterrupted supply of demineralized water is required, and supply the reactor pools when makeup water is needed.

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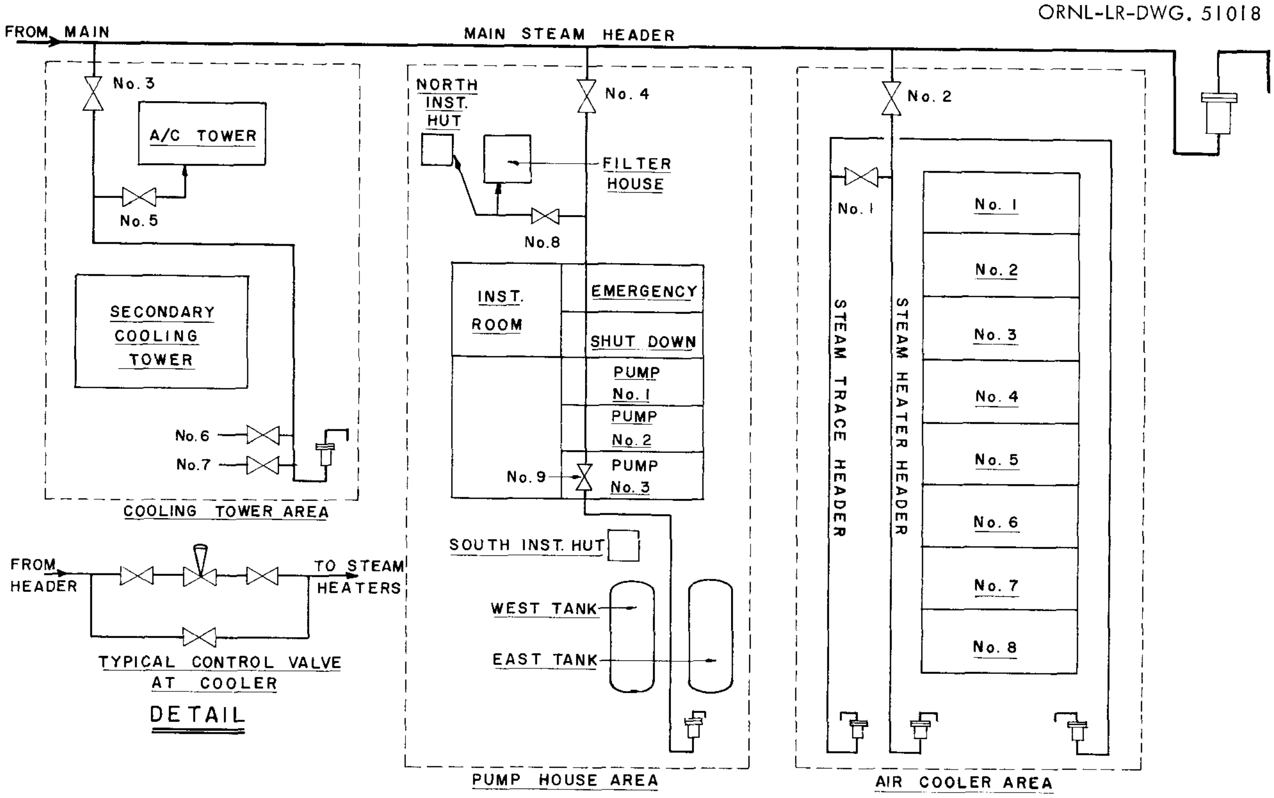

Fig. 6.11. Steam-Trace System.

Pool Makeup. – When water is lost from the pool system, the overflow to the side gutters, which feeds the reservoir tank, is reduced. Since the pool demineralizer pump takes its suction from this tank, the level in the tank will fall. Makeup is then supplied by a level-control valve, which supplies demineralized water directly from Building 3004. Without this makeup supply, sufficient water loss from the pool system through evaporation or other reasons could cause the demineralizer pump to lose suction, and the pump could be damaged.

Bottom-Plug Seals. – Valves and piping located in the subpile room route demineralized water from Building 3004 to a system of seals which prevent the radioactive water of the reactor cooling system from leaking past the bottom plug and shim-rod drives into the subpile room.

6.4.6 *Precautions Against Freezing*

The freezing of water in unsheltered portions of the ORR water system could lead to reactor shutdown, extensive damage to equipment, and costly maintenance. The prevention of freezing, therefore, is considered an essential part of operation.

Precautions against freezing in these areas include steam and electric heating and steam tracing. A flow diagram of the steam-trace system is shown in Fig. 6.11.

7. CORE PHYSICS, NUCLEAR DESIGN, AND HEAT TRANSFER

7.1 Introduction

The main purpose of the design of the ORR is to improve upon other reactor designs by providing increased accessibility to the core lattice. Many important features of other research reactors are also included in the design, but a number of important concepts in the arrangement of the reactor and its experimental facilities make the ORR an important and invaluable research and testing facility.

One of the unique features of the ORR is the location of the reactor tank in a pool of water. The water provides the necessary shielding for working above the reactor core and also makes access to the reactor core as convenient as it is in low-power pool-type reactors. The majority of experimenters desire to place their experimental rigs in the very-high-flux regions in the core and the reflector. Hence, easy access to the core was stressed during the design. To increase access to the core, the control-rod drive mechanisms are located below the core, and the upper grid plate which is usually present in this type of reactor has been eliminated. This last feature makes it possible to leave experiments in the core region while the reactor is being refueled.

Another factor which influenced the design of the ORR was the fact that there are other research reactors at ORNL which provide experimental facilities where low neutron fluxes are available. Thus, no special provisions were made to include experimental facilities wherein the fluxes were significantly below 10^{13} neutrons cm^{-2} sec^{-1}.

The reactor core is a heterogeneous type which uses enriched uranium fuel in the form of aluminum-clad aluminum-uranium alloy fuel plates. A fuel element consists of an assembly of 19 of the fuel plates. Demineralized light water serves as the reactor coolant and also the moderator. The reflector is composed of an arrangement of beryllium pieces which are physically interchangeable with each other and with the fuel elements. The thickness of the reflector therefore varies according to the particular fuel and experiment arrangement in the core. About 4 ft of water surrounds the reflector pieces. Figure 1.3 shows a cutaway view of the reactor core.

A rectangular aluminum box surrounds the core, and a grid plate is located below it to provide for the spacing and support of the fuel elements. Fuel elements, control rods, experimental rigs, and reflector elements are positioned within a 9 by 7 array in the grid plate.

The reactor is controlled by vertically positioning the control rods, which are located in the fuel and reflector regions of the reactor. Positions for up to 12 control rods are provided, but only six of these positions are used. The poison sections of the control rods are approximately as long as the height of the reactor core and consist of an aluminum-jacketed cadmium sheet formed into a rectangular box. The lower half of the rod consists of fuel of the same composition and general type as the regular fuel elements, except that only 14 fuel plates are included in the assembly. The control rods may be used as combination shim-safety-regulating rods, and the follower sections may be chosen according to the amount of fuel, poison, or reflector material desired, so that a specifically desired flux distribution may be achieved.

The heat transfer calculations for the ORR were originally performed considering initial operation of the reactor at 20 Mw and possible future operation at the present power level of 30 Mw. Coolant flow requirements were established as 12,000 gpm for 20-Mw operation and 18,000 gpm for 30-Mw operation. A conservative design criterion was used which assumed the simultaneous occurrence of the worst possible mechanical, nuclear, hydraulic, and thermal conditions in the same physical location. Data from operating experience of the Materials Testing Reactor (MTR) were used as an aid in establishing criteria and as a basis for comparison of calculations with experimentally observed quantities. Hydraulic tests were performed on fuel elements, control rods, and reflector elements as an integral part of the design effort.

The following sections describe the nuclear and heat transfer characteristics of the core and its various constituents.

7.2 Selection of Reactor Type and Materials

Consideration of the research facilities available at the Laboratory before the construction of the ORR indicated that several types of basic research facilities were urgently needed. Some of the most important needs were facilities that provided intense neutron beams, permitted tests of liquid-fueled reactor components, increased the Laboratory's radioisotope production capabilities, and provided additional convenient in-core irradiation sites. As an aid toward meeting these needs, the design and construction of the ORR was begun.

Investigations of various research reactor types, while bearing in mind the research facility needs, led to the choice of a reactor which was a modified version of the MTR. It was felt that a reactor of this type could be constructed to provide fast and thermal neutron fluxes in excess of 10^{14} neutrons cm^{-2} sec^{-1} while operating at power levels up to 30 Mw.

A lattice-type core fueled with plate-type elements was selected. A schematic diagram of the lattice arrangement of the core is shown in Fig. 7.1; Figs. 5.3 and 5.4 show the basic design of the fuel elements.

To keep research and development costs at a minimum, designs of fuel elements, beryllium reflector elements, and control rods similar to those used in the MTR were selected for use in the ORR. Aluminum-clad uranium-aluminum alloy fuel plates, light-water coolant and moderator, and beryllium reflector elements were chosen on the basis of the existence of considerable knowledge and experience in their fabrication and use. With regard to nuclear, heat transfer, economic, and practical considerations, the materials for these components were satisfactory and in some cases superior to other materials considered.

7.3 Nuclear Design

7.3.1 General Characteristics

The overall nuclear design of the reactor core and the ancillary experimental facilities was based mostly on the desire to achieve a high neutron flux while maintaining a high degree of accessibility and flexibility in the experimental facilities. Some of the typical characteristics of the ORR core have been discussed in Sect. 5.1. It should be emphasized that quantities such as fuel loading and reflector thickness may change from cycle to cycle, so the nuclear characteristics may vary accordingly.

The selection of the optimum core arrangement for each fuel cycle is made partly on the basis of the neutron-flux magnitudes and spectra desired in the various experimental facilities. By specifically arranging the fuel elements and/or reflector elements, numerous variations of the flux can be achieved. Neutron leakage from the fuel region through the reflector region supplies the neutron current desired for the experimental facilities such as beam holes and engineering facilities which are external to the core. The leakage from the core to those facilities depends on the core geometry and size, its metal-to-water ratio, the power distribution, and the parasitic absorption.

The core volume was determined by heat transfer requirements. Variations in the positions of fuel elements, reflector elements, fuel and fission product concentration in the fuel elements, and mechanical assembly characteristics naturally cause significant variations and uncertainties in the power distribution. Thus, considerable conservatism was employed in the design of the fuel elements and in establishing the core volume.

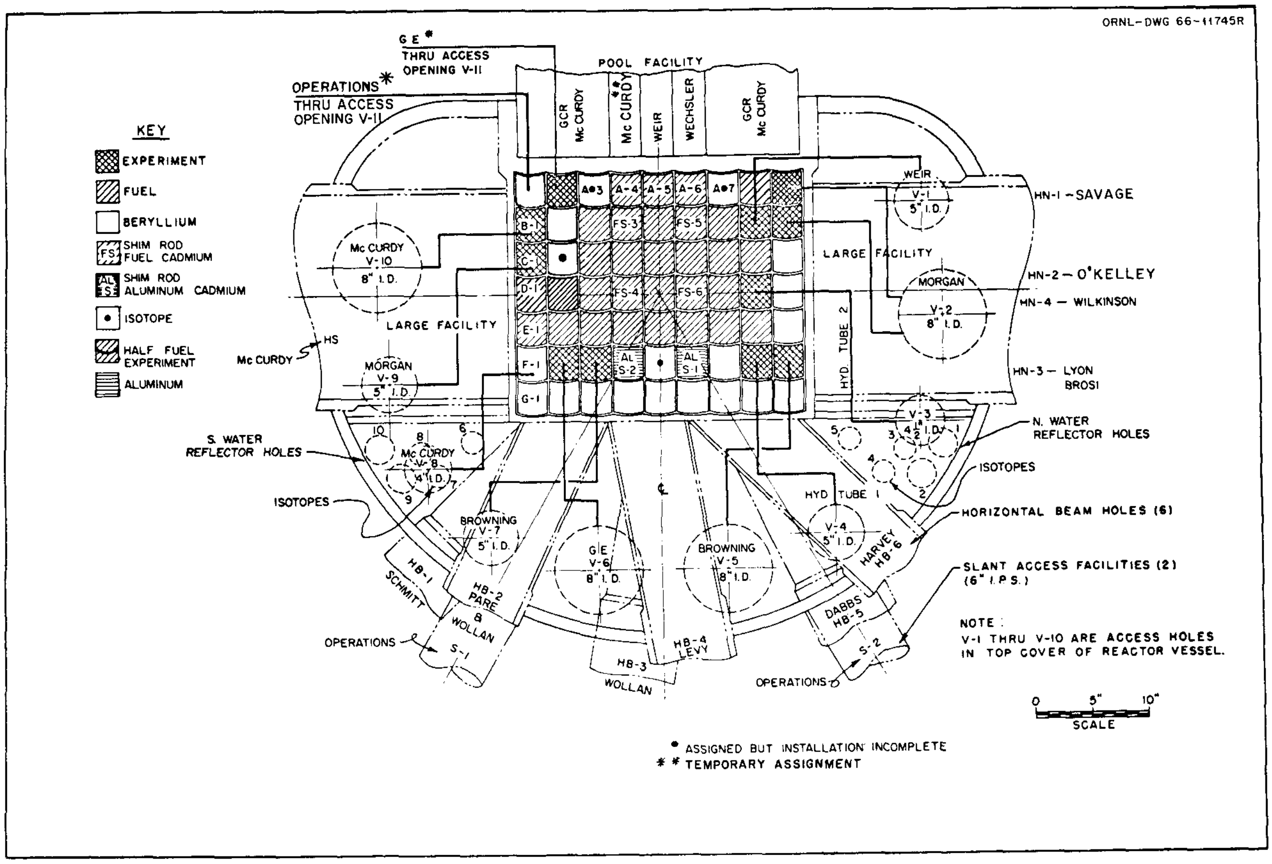

Fig. 7.1. Lattice Arrangement of the ORR Core.

7.3.2 Fuel Loadings and Neutron Flux Distributions

The fuel loading pattern of the ORR is very flexible and within certain practical limits may be varied considerably. The relatively large fuel and reflector grid provides space for numerous in-reactor experiments while still maintaining a large fuel region. Various fuel loadings have been used successfully, their configurations varying from a simple 3 by 3 square array to a more complex one like that shown in Fig. 7.1. Two main criteria are employed in selecting the fuel loadings. First, the amount of fuel added at each refueling is limited to a value such that the reactor will not become critical until the shim rods have been withdrawn approximately halfway. This seemingly arbitrary criterion is a compromise between an effort to achieve as long an operating cycle as possible and a desire to limit the excess reactivity to a value which would allow adequate control should some system malfunction occur. Second, the lighter partially burned elements are placed near the center of the core, and the heavier newer elements are positioned around the periphery of the fuel region. This latter criterion has a twofold purpose. Placing the partially burned elements toward the center of the core in a region of high flux increases the fraction of burnup in those elements. In addition, placing the elements containing more fuel in the outer positions, where the flux is somewhat lower, decreases the possibility of high power density and, consequently, of hot spots occurring in those elements.

Criticality calculations were performed[1] for several core configurations and for various fuel concentrations. The calculated critical mass for a 3 by 3 core was 1150 g, and the experimentally

[1]F. T. Binford, *Two Group Calculations for the Flux Distribution and Critical Mass in Clean Cold ORR Cores*, ORNL-CF-57-5-31 (Jan. 5, 1959).

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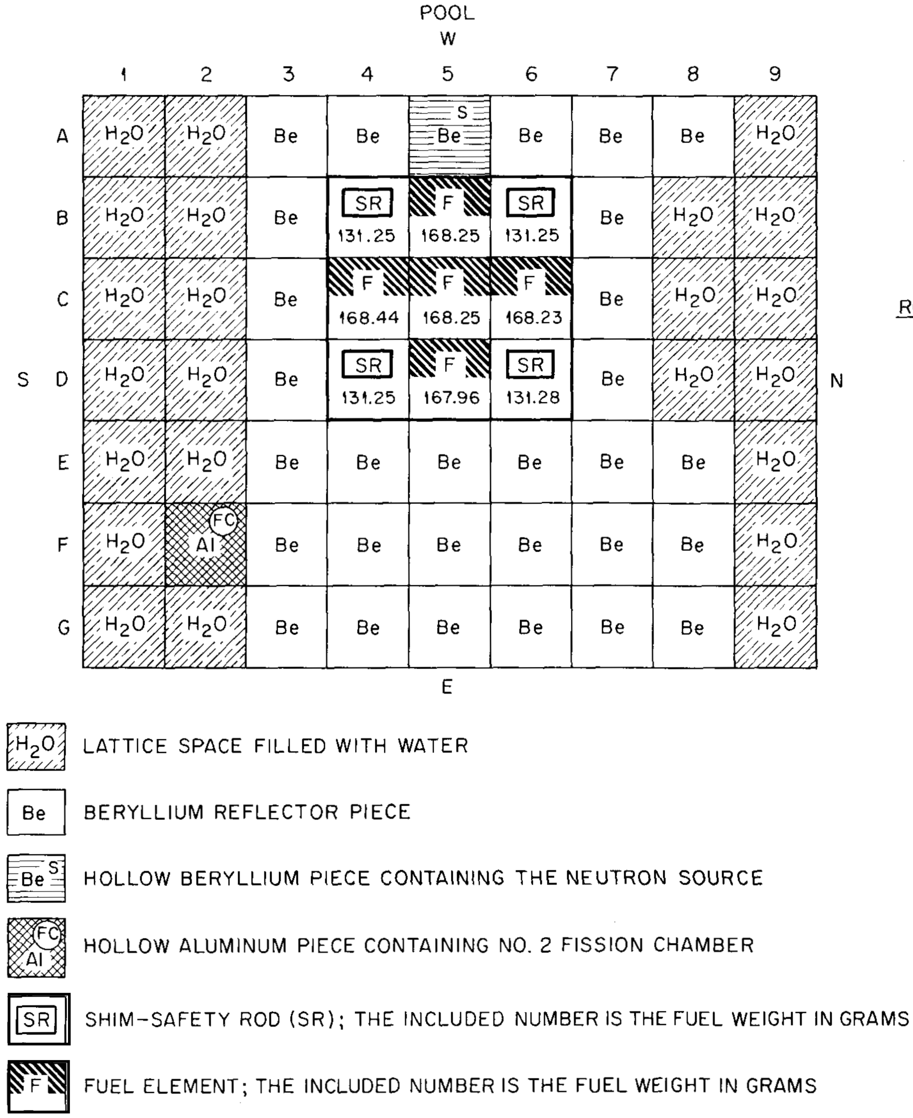


ROD POSITIONS AT CRITICAL

ROD NO.	POSITION	INCHES OUT
3	D-6	25.0
4	D-4	25.0
5	B-6	24.5
6	B-4	24.5

Fig. 7.2. Diagram of First Critical Core.

determined value[2] was approximately 1280 g. Figure 7.2 is a diagram of the first critical core. Results of the criticality calculations are indicated in Table 7.1 and are plotted graphically in Figs. 7.3 and 7.4.

Four of the core configurations used in the initial nuclear tests, along with some of the pertinent nuclear data, are shown in Figs. 7.5 through 7.8. These configurations may be compared with one more recently used that is shown in Fig. 7.1.

There is no characteristic detailed neutron flux distribution for the ORR core, since the fuel and/or reflector loading and configuration may change from cycle to cycle. It therefore is meaningful to present a flux pattern in a certain position in the core or reflector only if the complete core description is also given. However, in certain cases it may be informative to describe some of the flux distributions, because from these distributions some very general observations can be made about the

[2]C. D. Cagle and R. A. Costner, Jr., *Initial Post-Neutron Measurements in the ORR*, ORNL-2559 (May 28, 1959).

Table 7.1. Values Obtained for K_{eff}

Core Type[a]	Fuel Loading (kg)	Fuel Concentration (g/element)	K_{eff}	K_{∞}	L^2
4 × 7	2.30	82.1	0.9988	1.3536	5.4968
	3.00	107.1	1.1043	1.4781	4.6015
	3.50	125.0	1.1621	1.5447	4.1220
	4.00	142.8	1.2096	1.5988	3.7329
4 × 6	2.00	83.3	0.9932	1.3620	5.4517
	2.50	104.2	1.0819	1.4670	4.6975
4 × 5	1.60	80.0	0.9565	1.3407	5.5900
	2.50	125.0	1.1284	1.5447	4.1220
	3.00	150.0	1.1929	1.6177	3.5971
	3.50	175.0	1.2441	1.6741	3.1909
4 × 3	1.20	100.0	0.9646	1.4469	4.8261
	1.50	125.0	1.0482	1.5447	4.1220
	2.00	166.7	1.1480	1.6568	3.3157
	2.50	208.3	1.2185	1.7321	2.7732
3 × 7	1.50	71.4	0.8999	1.2842	5.9969
	2.00	95.2	1.0156	1.4243	4.9884
	2.50	119.0	1.1007	1.5241	4.2703
3 × 5	1.00	66.7	0.8358	1.2491	6.2497
	1.50	100.0	0.9974	1.4469	4.8263
	2.00	133.0	1.1070	1.5714	3.9310
3 × 3	1.00	111.0	0.9524	1.4943	4.4857
	1.50	166.7	1.0961	1.6568	3.3158
4 × 6	2.5	104.2	0.9423	1.4670	4.6975
	3.0[b]	125.0	1.0004	1.5464	4.1266
	3.4	141.7	1.0381	1.5973	3.7609

[a]The first number indicates the dimension in the west-east direction; the second number indicates the dimension in the south-north direction. All cores are beryllium reflected except where otherwise noted.

[b]Water reflected.

fluxes in other core configurations. Furthermore, in some of the core positions no gross change in the flux distribution will occur, but the magnitude and/or the spectra will vary slightly.

For 30-Mw operations, the average[3] thermal neutron flux in the ORR core is 1.40×10^{14} neutrons cm^{-2} sec^{-1}, the average epithermal flux is 3.5×10^{14} neutrons cm^{-2} sec^{-1}, and the maximum thermal neutron flux is 5×10^{14} neutrons cm^{-2} sec^{-1}. These values, of course, vary according to the fuel loading; this variation is illustrated in Fig. 7.9.

Numerous flux distribution measurements have been made in the ORR, but usually they were performed under special circumstances to determine effects of various components on that distribution or to obtain information about a new component. However, some measurements were general and are of sufficient interest to warrant their presentation here. A few measurements are presented to indicate the variations which may occur when some fuel and/or reflector changes are made. Figures 7.10 through 7.12 show the longitudinal thermal flux distributions for some of the positions in row

[3]This average is computed as a mean value over the volume of the core and over the duration of the fuel cycle.

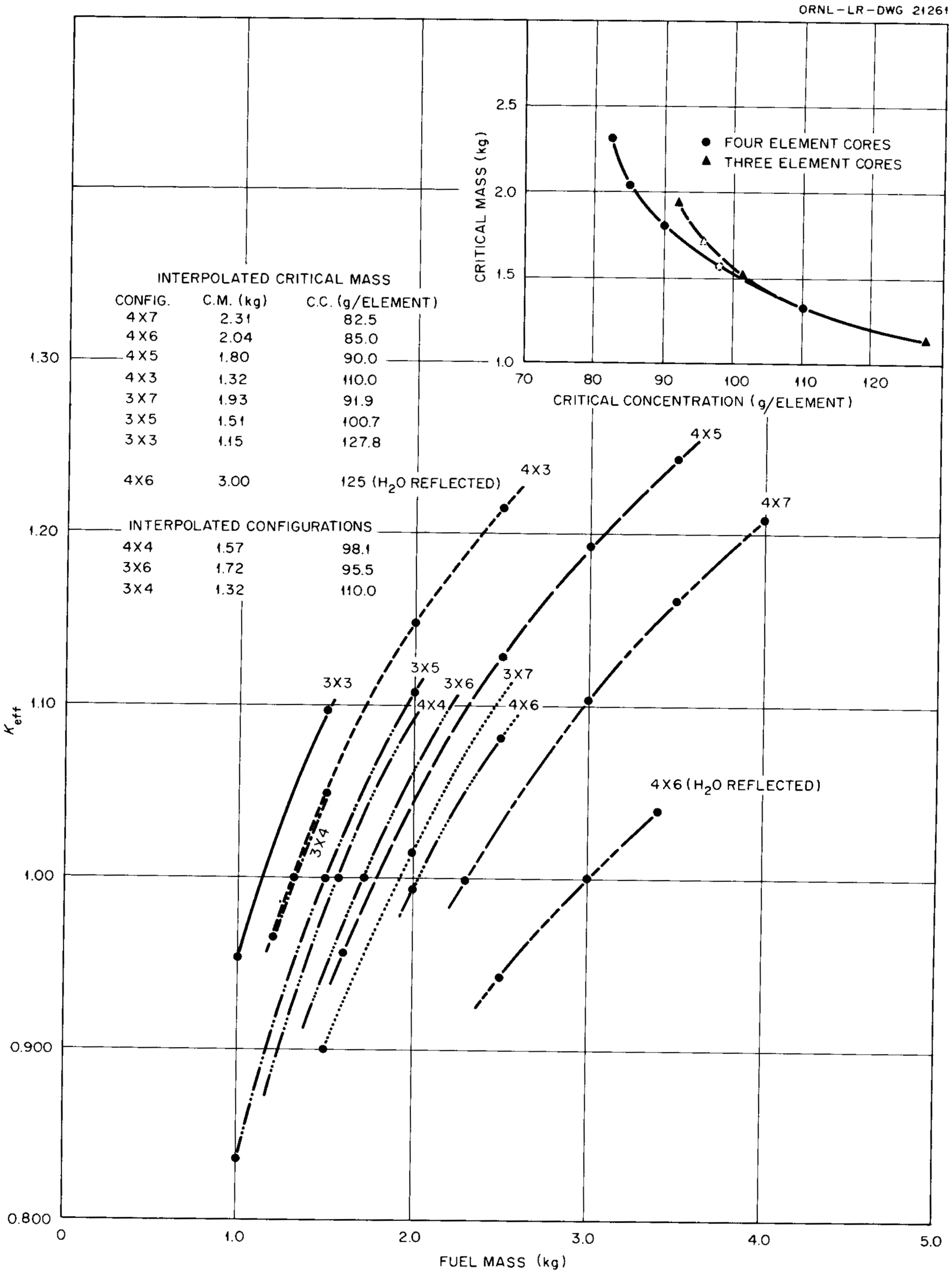


Fig. 7.3. Plot of K_{eff} vs Fuel Mass for Various ORR Core Configurations.

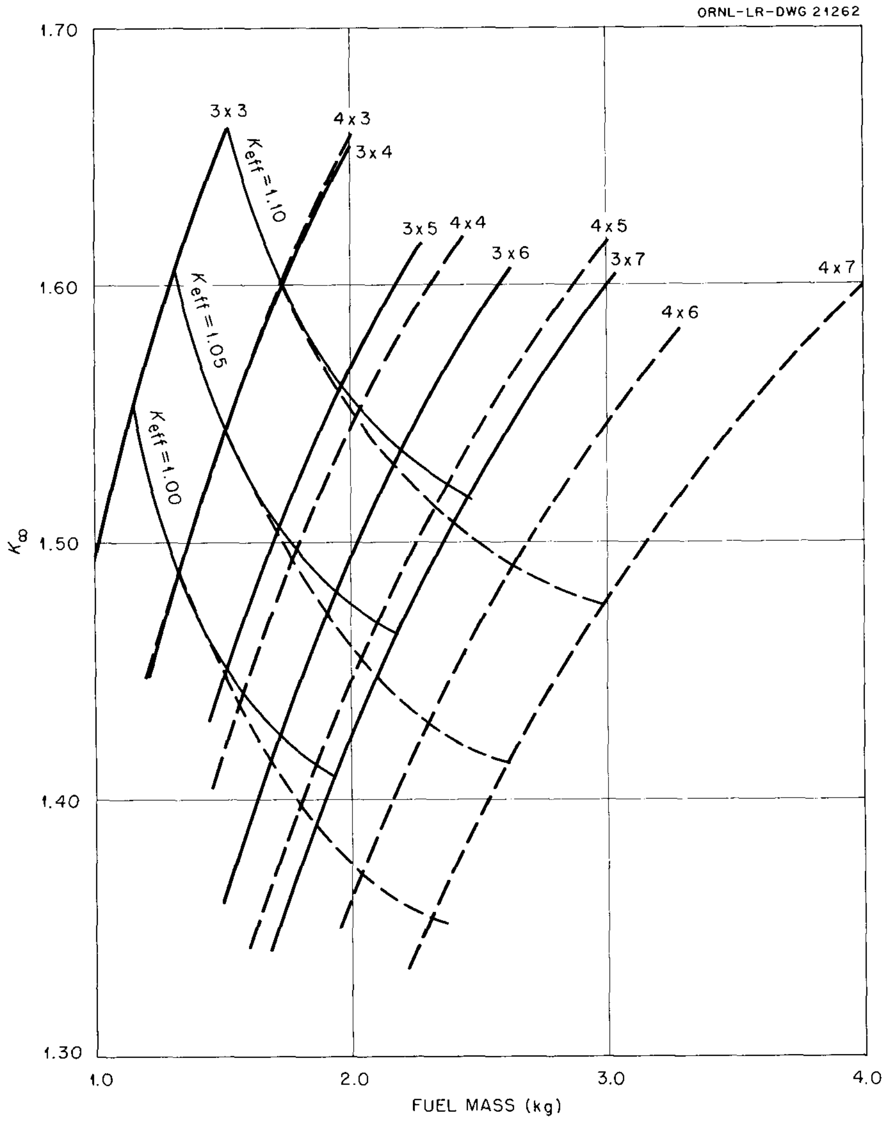
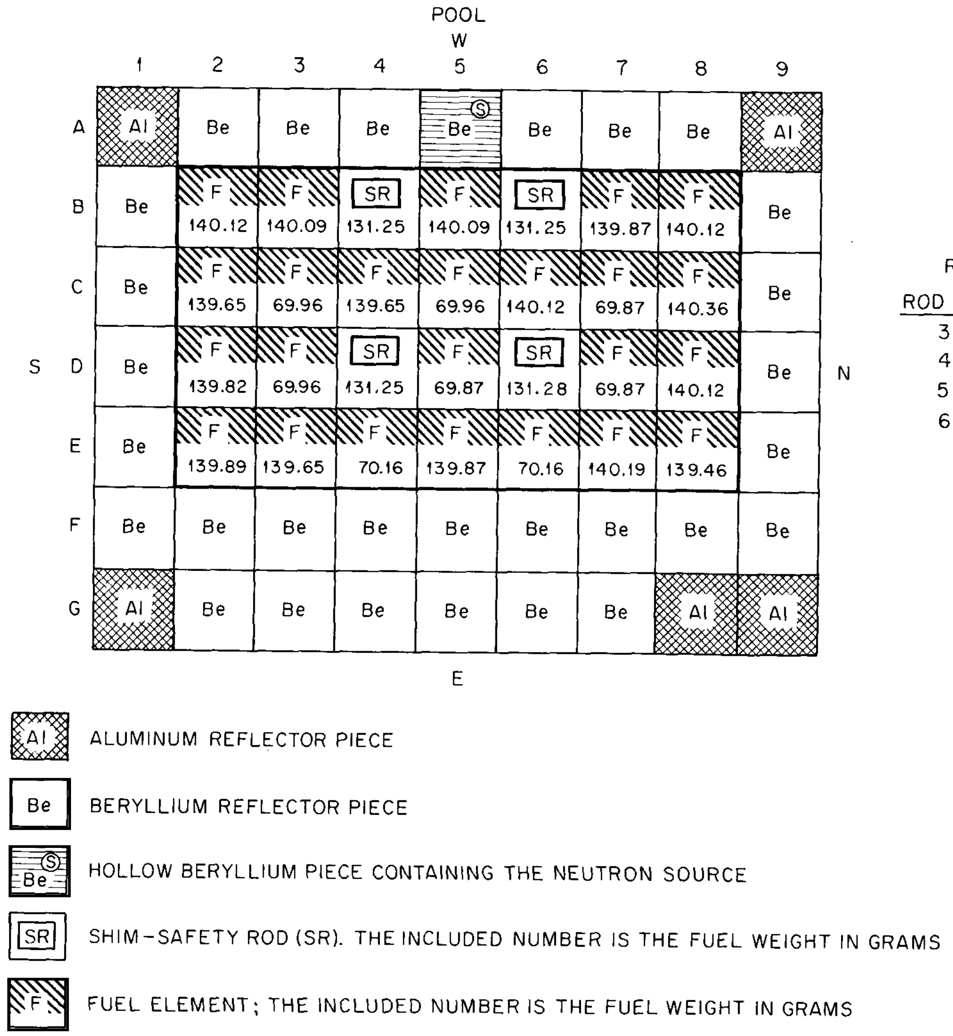


Fig. 7.4. Plot of K_∞ vs Fuel Mass Showing Lines of Constant K_{eff}.

G for the core shown in Fig. 7.8. Other similar measurements for the same core are presented in ref. 2. The ratio $\Phi/\bar{\Phi}$ as used here is the ratio of the thermal flux at that point to the average thermal flux in the core, and the curves labeled with and without experiments are self-explanatory. Flux distributions for the core loading given in Fig. 7.13 are shown in Figs. 7.14 through 7.16. Other longitudinal flux distributions are available[2,4] for these and other core loadings, but these particular measurements lend understanding to the general patterns and characteristics of those distributions. One noteworthy point is that in all cases the flux distributions peak slightly below the center of the core, resulting in a skewed effect. This effect may be attributed to the positions of the control rods, since their poison sections are usually located in the upper half of the core. The skewed appearance tends to decrease during the fuel cycle as burnup continues and, consequently, as the rods are further withdrawn. However, this distortion will not completely disappear or reverse itself toward the upper end, because the fuel distribution will have become distorted, thereby leaving a higher fuel concentration toward the tops of the

[4] J. A. Cox, *ORR Operations for Period April 1959 to April 1960*, ORNL-CF-60-9-2, pp. 22–36 (Sept. 30, 1960).

ORNL–LR–DWG 35170



ROD POSITION AT CRITICAL

ROD NO.	POSITION	INCHES OUT
3	D–6	15.43
4	D–4	15.42
5	B–6	15.42
6	B–4	15.41

Fig. 7.5. ORR Core Configuration.

fuel elements. The fuel elements for the ORR are built specifically so that they may be installed with either end up. Ideally, therefore, the fuel elements could be reversed periodically, so that the distortion in the fuel burnup could be reduced somewhat. Figure 7.17 shows the flux distribution along three coolant channels of a shim-rod follower with a 131-g ^{235}U fuel loading. Relative flux here indicates normalization to the maximum measured flux. Figures 7.18 and 7.19 show a normalized empirically determined flux distribution in fuel elements and the followers of fuel shim rods.

Horizontal flux distributions are affected more by fuel loadings than longitudinal or vertical ones. A flux traverse along the center plane of the fourth column of the core of Fig. 7.20 is shown in Fig. 7.21. These measurements were taken at a position 16 in. below the top of the fuel plates at the approximate location of the maximum flux. A set of measurements made on the core of Fig. 7.8 along row *E* is shown in Fig. 7.22. More detailed measurements of the distributions across individual fuel elements have been performed.[5]

A general view of the flux patterns in the core of Fig. 7.13 is shown in Fig. 7.23, where the average element flux is compared with the average core flux.

[5]J. A. Cox, *ORR Operations for Period April 1960 to April 1961*, ORNL-TM-10, pp. 62–70 (Oct. 20, 1961).

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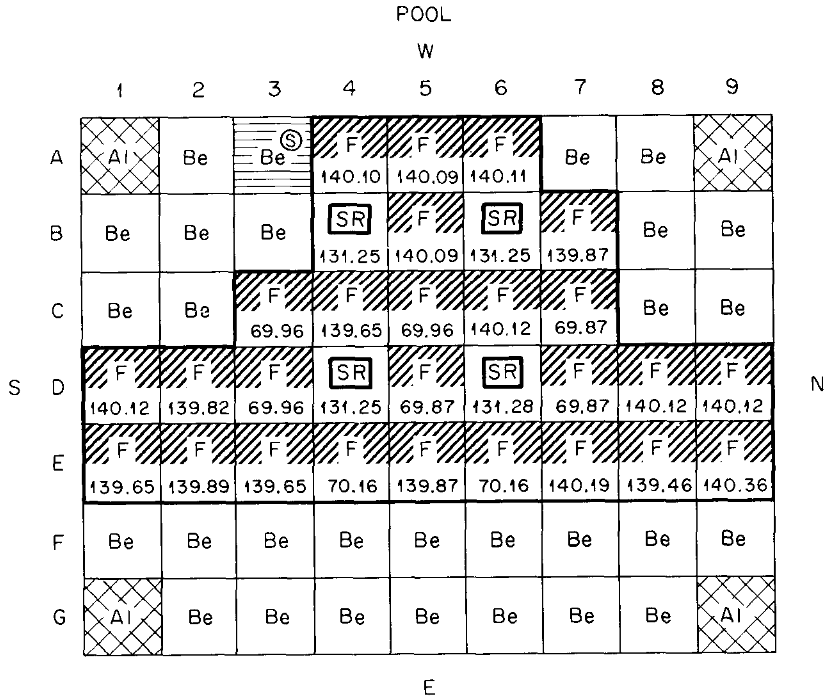


ROD POSITIONS AT CRITICAL

ROD NO.	POSITION	INCHES OUT
3	D-6	16.01
4	D-4	16.01
5	B-6	16.00
6	B-4	16.00

Al — ALUMINUM REFLECTOR PIECE

Be — BERYLLIUM REFLECTOR PIECE

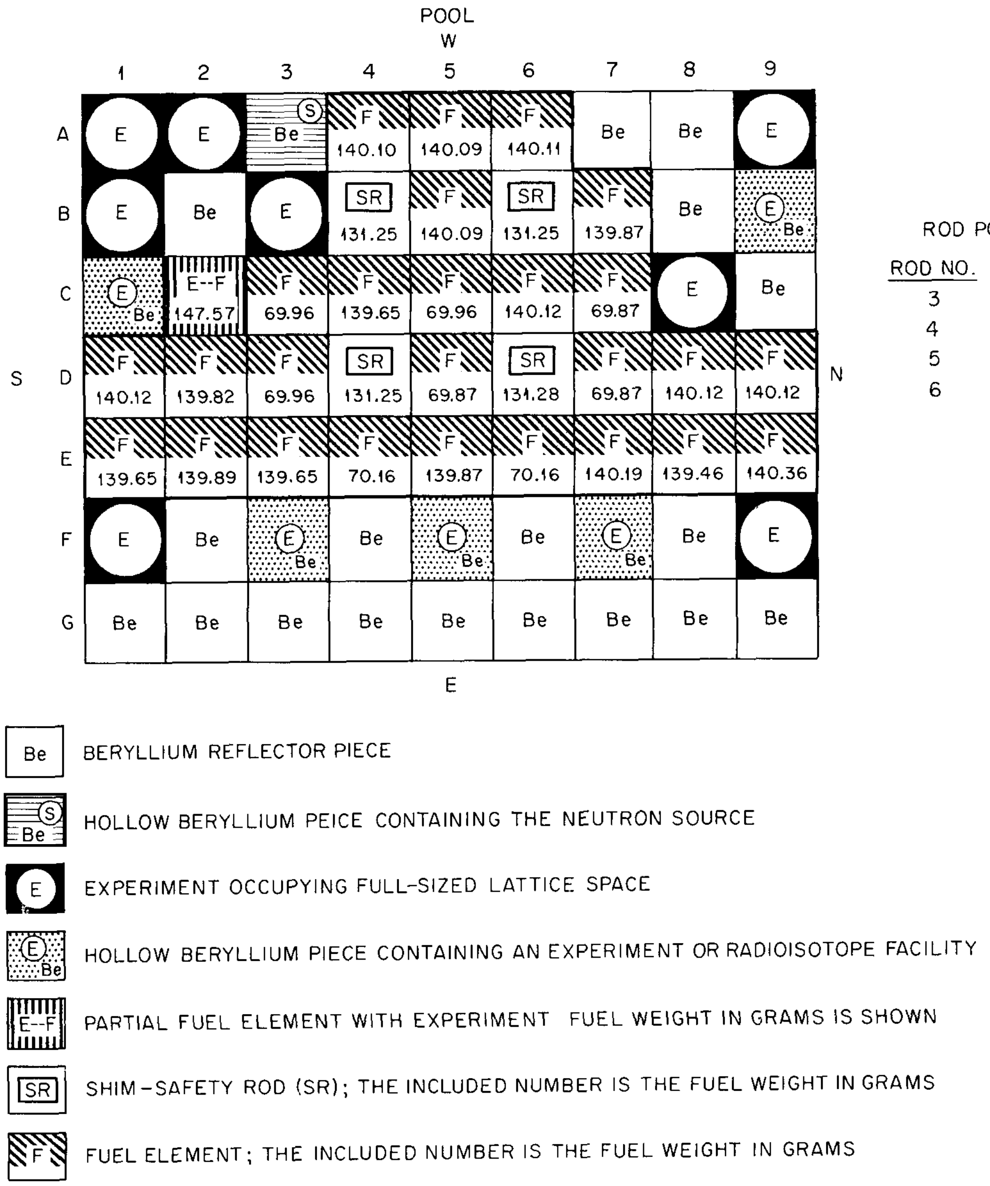
Be Ⓢ — HOLLOW BERYLLIUM PIECE CONTAINING THE NEUTRON SOURCE

SR — SHIM-SAFETY ROD (SR); THE INCLUDED NUMBER IS THE FUEL WEIGHT IN GRAMS

F — FUEL ELEMENT; THE INCLUDED NUMBER IS THE FUEL WEIGHT IN GRAMS

Fig. 7.6. ORR Core Configuration.

ORNL-LR-DWG 35172

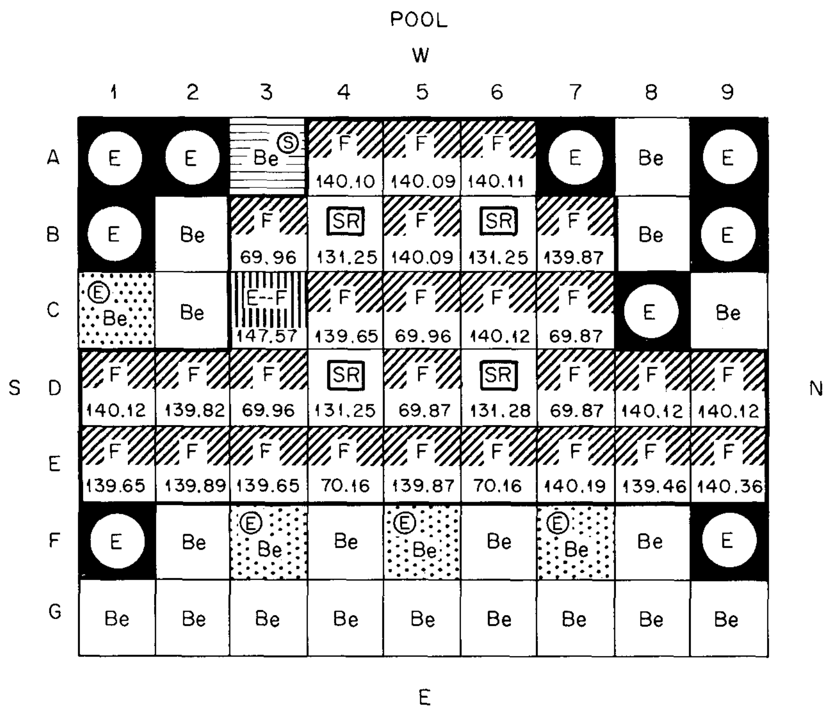


ROD POSITION AT CRITICAL

ROD NO.	POSITION	INCHES OUT
3	D-6	19.21
4	D-4	19.21
5	B-6	19.21
6	B-4	19.21

Fig. 7.7. ORR Core Configuration.

ORNL—LR—DWG 35173



ROD POSITION AT CRITICAL

ROD NO.	POSITION	INCHES OUT
3	D-6	18.14
4	D-4	18.14
5	B-6	18.14
6	B-4	18.14

Be — BERYLLIUM REFLECTOR PIECE

Be S — HOLLOW BERYLLIUM PIECE CONTAINING THE NEUTRON SOURCE

E — EXPERIMENT OCCUPYING FULL-SIZED LATTICE SPACE

E Be — HOLLOW BERYLLIUM PIECE CONTAINING AN EXPERIMENT OR RADIOISOTOPE FACILITY

E--F — PARTIAL FUEL ELEMENT WITH EXPERIMENT; FUEL WEIGHT IN GRAMS IS SHOWN

SR — SHIM-SAFETY ROD (SR); THE INCLUDED NUMBER IS THE FUEL WEIGHT IN GRAMS

F — FUEL ELEMENT; THE INCLUDED NUMBER IS THE FUEL WEIGHT IN GRAMS

Fig. 7.8. ORR Core Configuration.

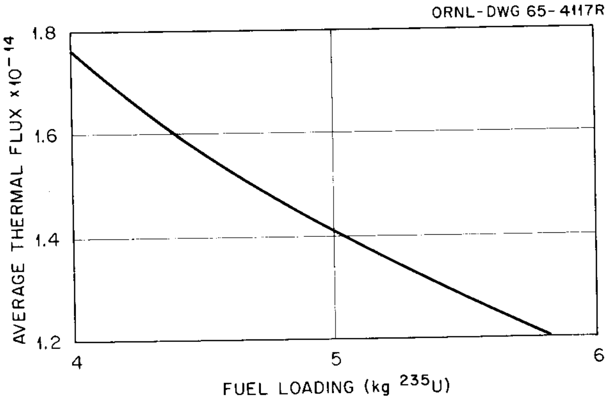


Fig. 7.9. Flux vs Fuel Loading in the ORR Core.

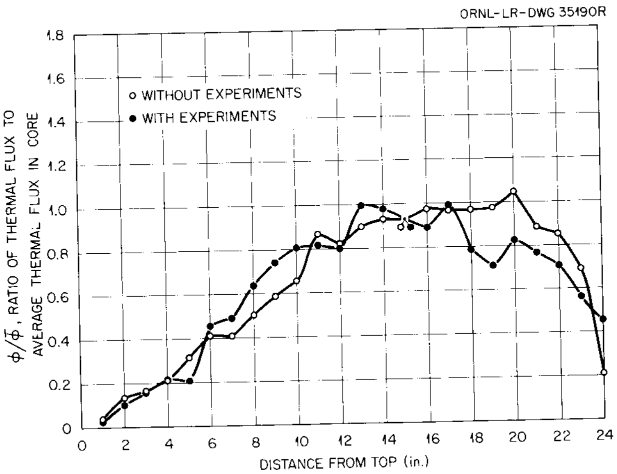


Fig. 7.10. Flux Distribution in the ORR Core.

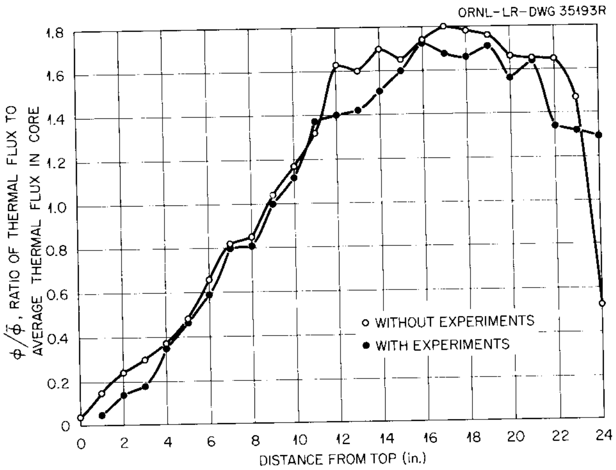
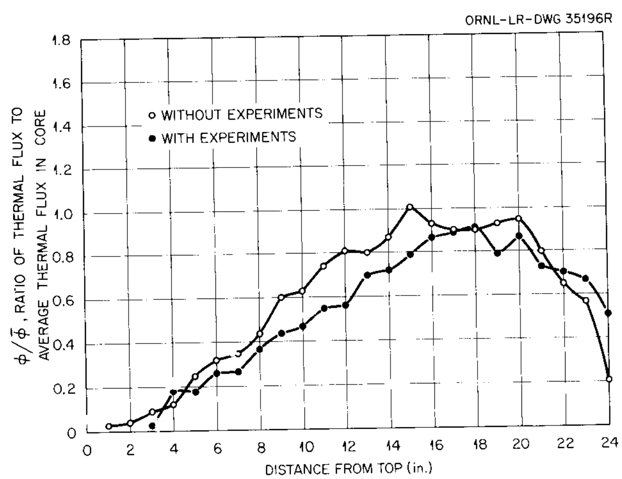


Fig. 7.11. Flux Distribution in the ORR Core.

Fig. 7.12. Flux Distribution in the ORR Core.



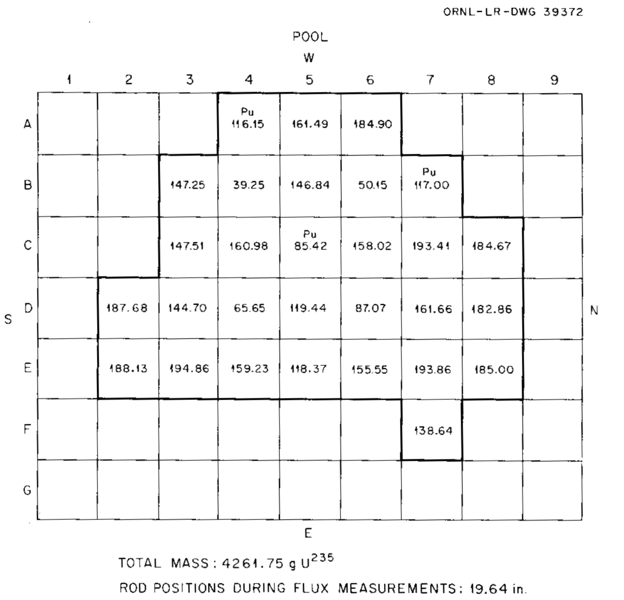


Fig. 7.13. ORR Cycle 12a Core.

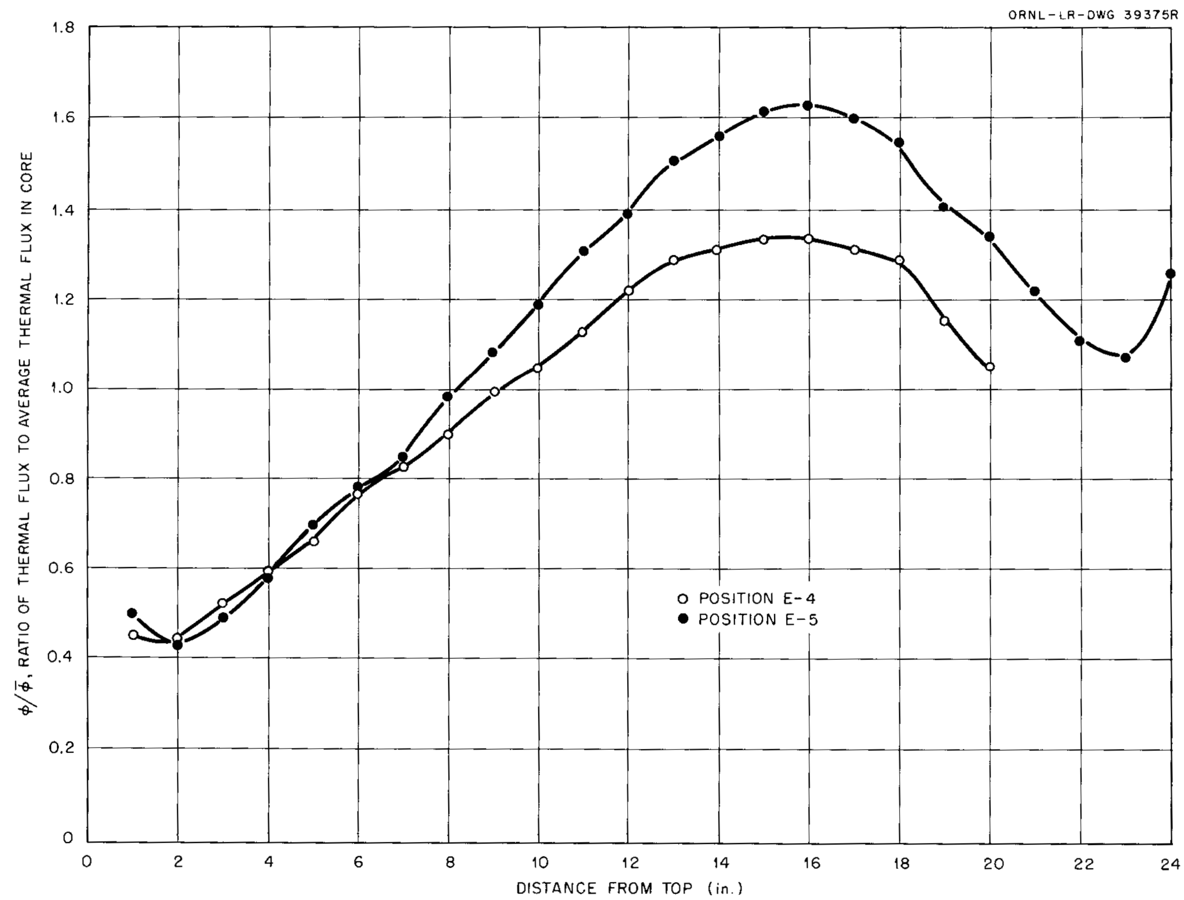


Fig. 7.14. Sample Flux Traverse.

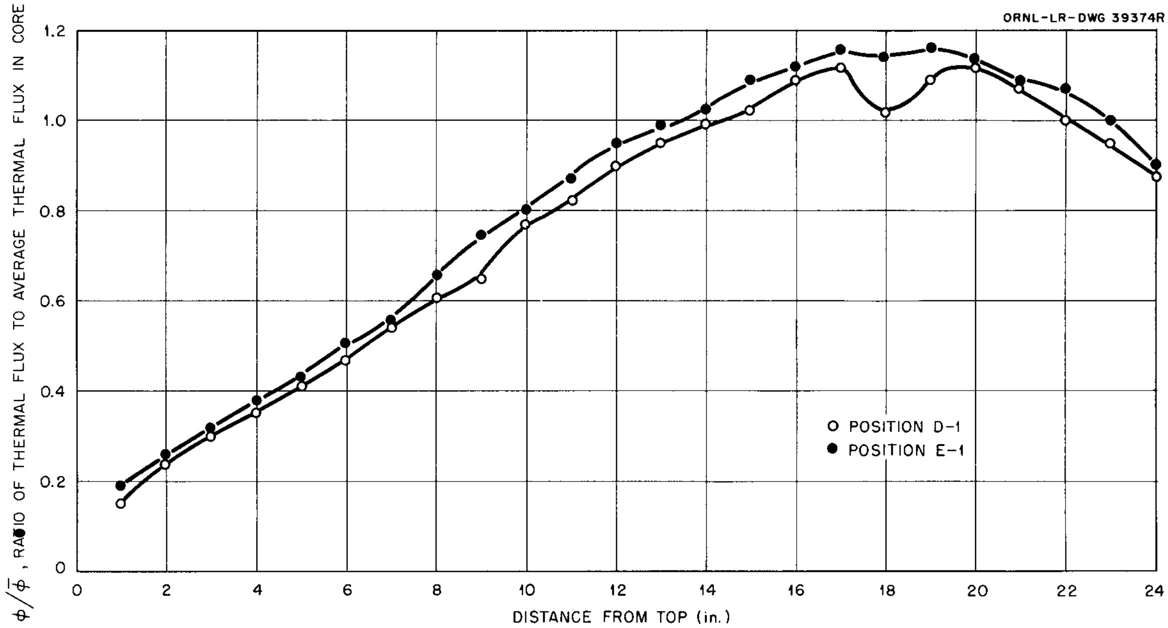


Fig. 7.15. Sample Flux Traverse.

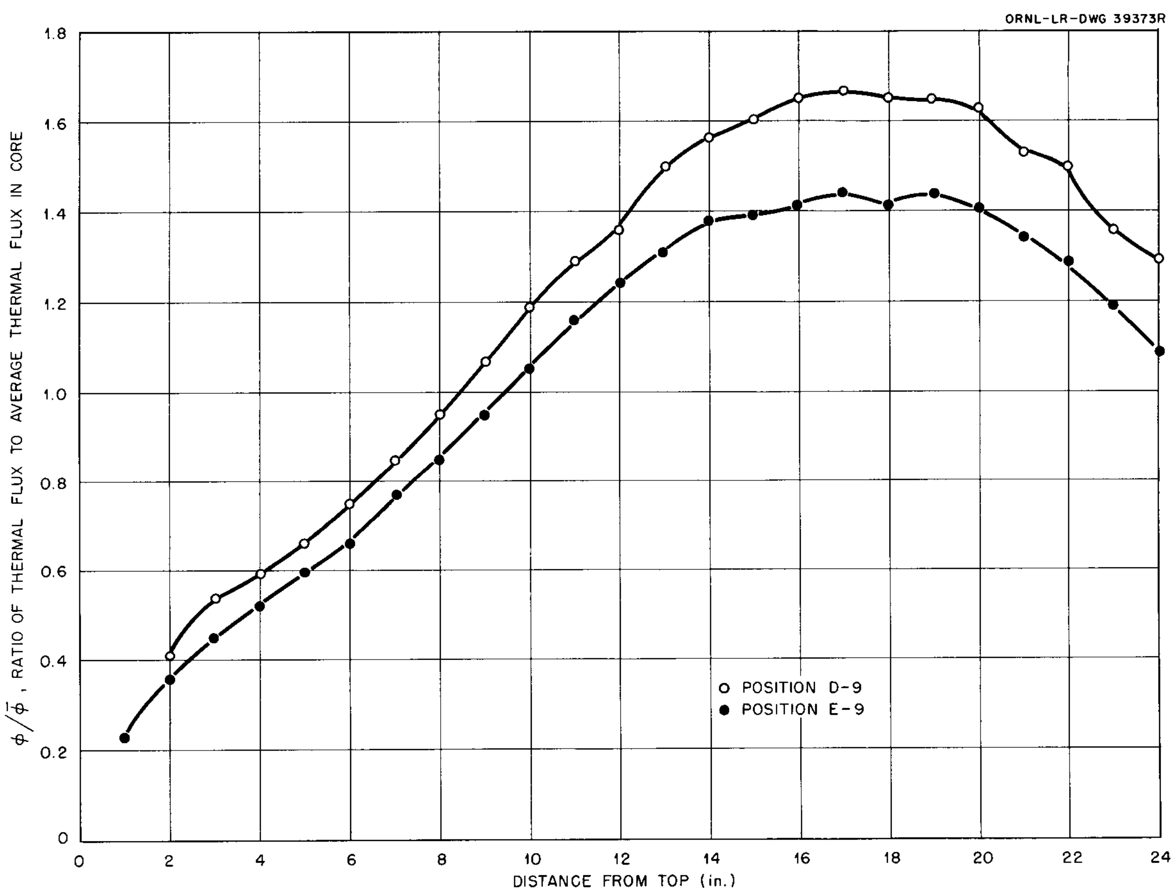


Fig. 7.16. Sample Flux Traverse.

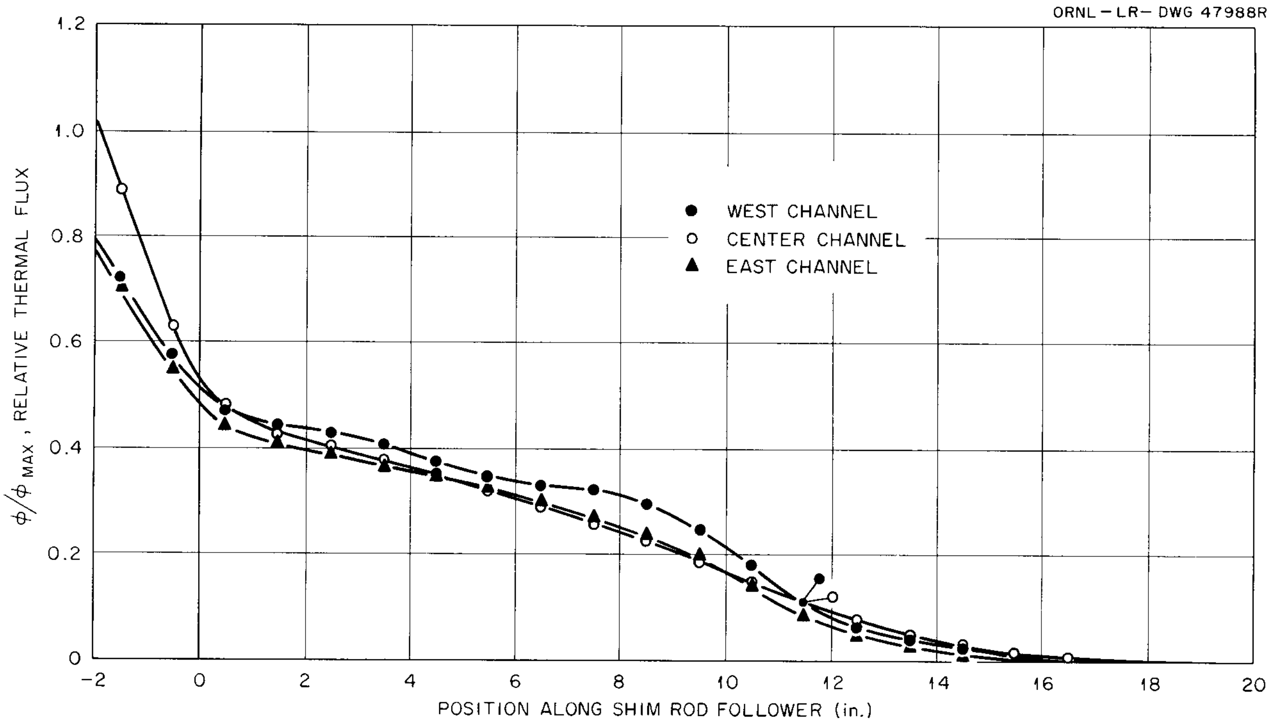


Fig. 7.17. Relative Fluxes in Different Channels Along the 131-g Fuel Shim-Rod Followers.

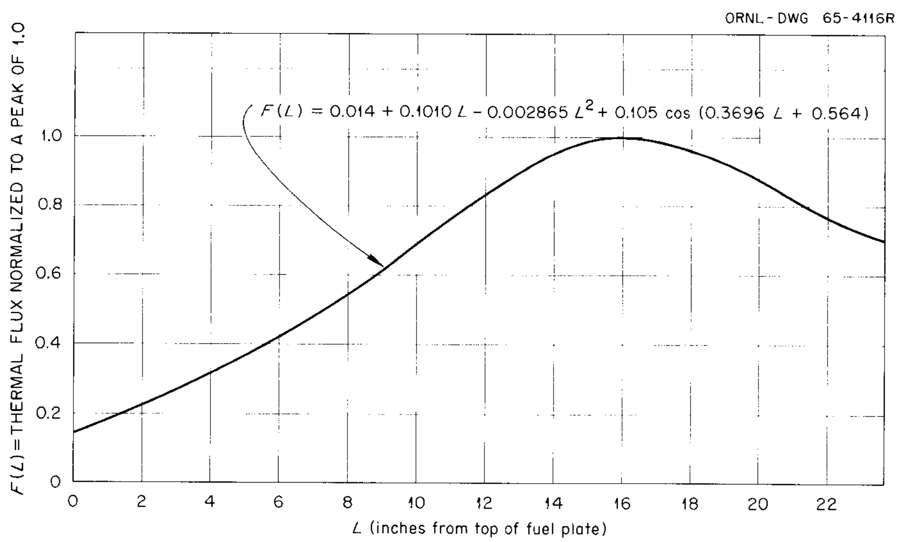


Fig. 7.18. Flux Distribution in an ORR Fuel Element.

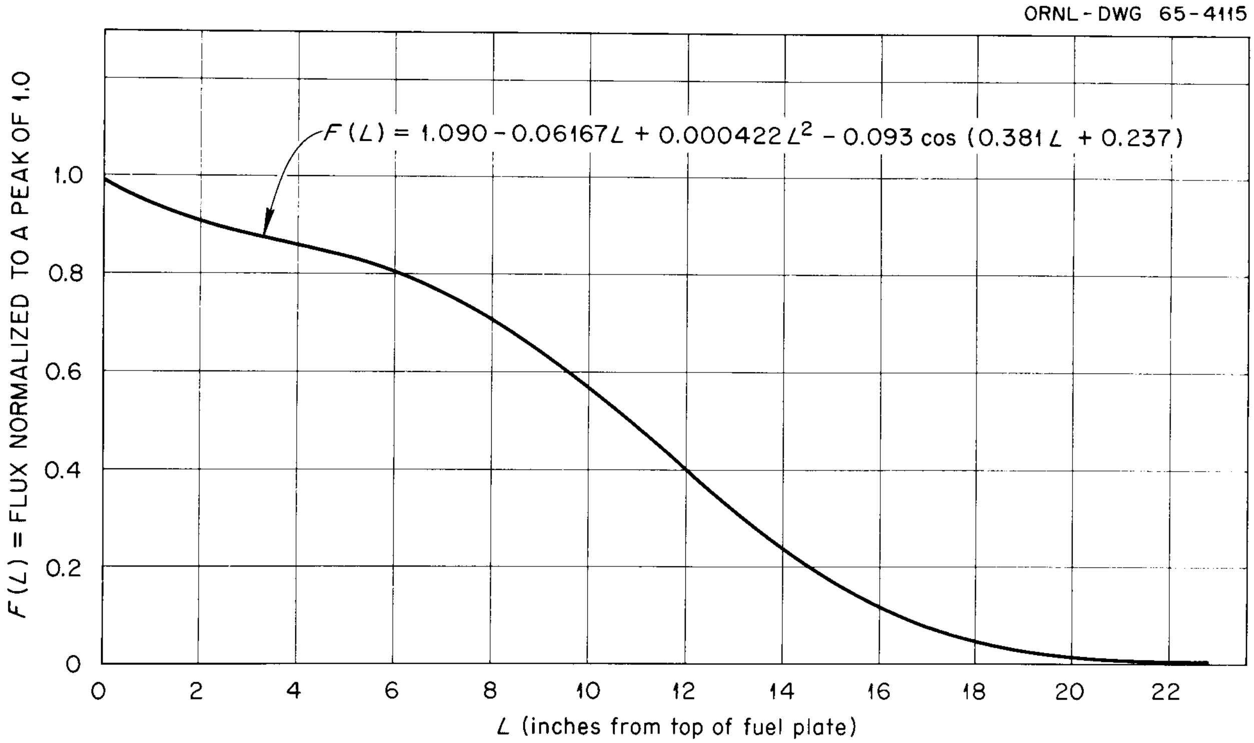


Fig. 7.19. Flux Distribution in an ORR Shim-Rod Fuel Follower.

ORNL-LR-DWG 51035R

	1	2	3	4	5	6	7	8	9
A	Be	Be	Be	F 163	F 169	F 166	Be	Be	Be
B	Be	Be	F 193	S 142	F 153	S 110	F 189	Be	Be
C	Be	Be	F 131	F 148	F 163	F 146	F 157	F 161	Be
D	Be	F 159	F 189	S 91	F 161	S 69	F 188	F 142	Be
E	Be	F 160	F 189	F 188	F 158	F 159	F 187	F 162	Be
F	Be	Be	Be	Be	Be	Be	F 148	Be	Be
G	Be	Be	Be	Be	Be	Be	Be	Be	Be

F – FUEL
S – SHIM ROD
Be – BERYLLIUM REFLECTOR

Fig. 7.20. Core Loading Pattern of ORR.

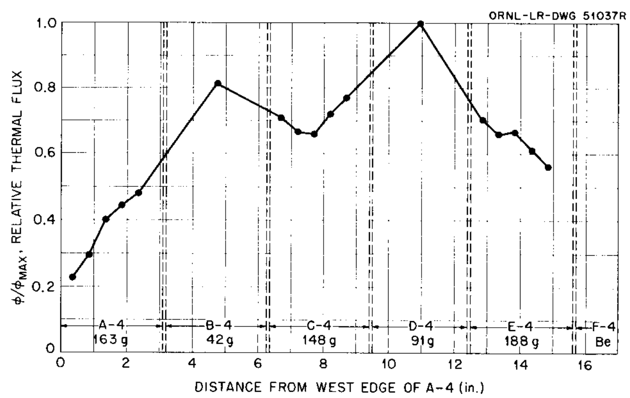


Fig. 7.21. Flux Traverse Along Center Plane of Column 4.

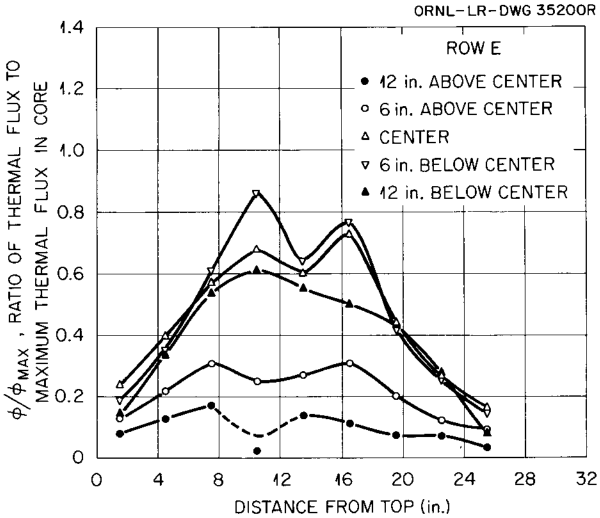


Fig. 7.22. Flux Traverse Along Row *E*.

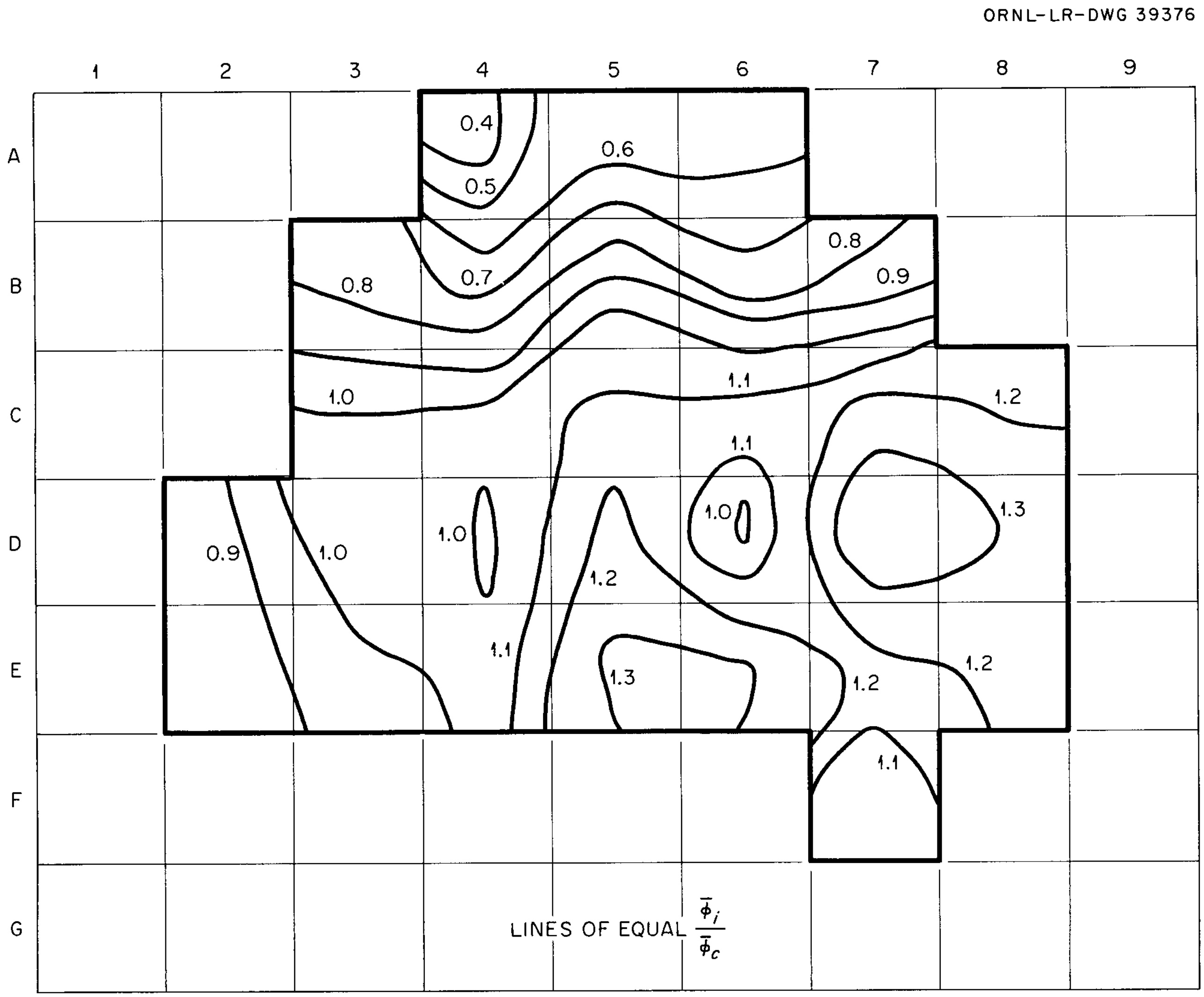


Fig. 7.23. Lines of Equal Ratio of Average Element Flux to Average Core Flux.

Regarding these flux distributions in the ORR, there are some important points to consider. Generally, when an element with a low fuel concentration is placed next to an element with a high fuel concentration, the flux will be peaked in the lighter element and depressed in the heavier one. This is to be expected, since to a first approximation the local flux varies inversely with the macroscopic absorption cross section. Hence, due to the lower cross section of the lighter element, the flux increases in that position. This idea may be extended quite analogously to new and partially burned fuel elements, because the new elements have a higher cross section and the burned elements have a lower cross section. In addition, in large water gaps such as those that exist in the transition region of shim rods with fuel followers, a flux-trapping effect may result, thereby considerably increasing the thermal flux in that general area.

7.3.3 Fuel Cycles

An operating cycle for the ORR consists of full power operation at 30 Mw for seven weeks and an end-of-cycle shutdown that lasts about one week. The seven weeks of operation is in turn divided into periods termed fuel cycles, which are periods during which no fuel or reflector changes are made. The reactor must be shut down on a fixed schedule to remove irradiated isotopes and to reload isotope target material. Therefore, an operating period corresponding to the optimum or most advantageous isotope irradiation time has been selected. This operating period has been chosen to be about four weeks following the beginning of an operating cycle, but it may be a few days more or less than this amount, depending on the length of the preceding end-of-cycle shutdown. However, the maximum allowable excess reactivity of a core loading is not sufficient to be able to operate the reactor at 30 Mw for more than about 20 days. Consequently, a representative operating cycle may be divided into fuel cycles similar to those shown in Table 7.2. Deviations from this schedule are usually due to unexpected shutdowns resulting from experimental difficulties and their subsequent control actions, reactor and experiment instrument malfunctions, reactor component malfunctions, etc. Most of the unscheduled shutdowns last only a few minutes, but they may last several hours.

Table 7.2. Representative Fuel Cycles

Fuel Cycle Designation[a]	Operating Period (weeks)	Reason for Termination
a	~ 2	Refueling
b	~ 2	Midcycle shutdown; isotope work and refueling
c	$\sim 1\frac{1}{2}$	Refueling
d	$\sim 1\frac{1}{2}$	End-of-cycle shutdown
	7	

[a]This designation follows operating cycle number (e.g., cycle 42a).

Because of the buildup of xenon after the reactor shutdowns, the reactor often has to be reloaded following these downtimes in order that it may be restarted within a reasonable time. If the shutdowns occur following prolonged operation of the ORR at 30 Mw, the majority of the centrally located elements must be replaced by other elements which will provide sufficient reactivity to start up and operate the reactor. If one of these reloadings must be performed 14 to 16 days before the midcycle or end-of-cycle shutdowns, then normally no additional refueling will be scheduled until that midcycle or end-of-cycle shutdown.

The operating criteria for the ORR specify that the ''excess reactivity loading above clean-cold-critical will not exceed that which will permit achievement of criticality with the rods withdrawn less than half their reactivity worths.'' Prior to each fuel cycle, the fully withdrawn positions of the ganged rods are determined, and the critical loading specification is taken to be one-half of that distance.

During the fuel cycle, reactivity variations are due mostly to fuel burnup and fission product accumulation. The most significant of the fission products is, of course, ^{135}Xe, which reaches its steady state or equilibrium value approximately two days after startup. This value, assuming operation at 30 Mw with an average thermal neutron flux of 1.40×10^{14} neutrons cm^{-2} sec^{-1}, is about $-0.047\ \Delta k/k$. However, in the event of a reactor shutdown, the xenon concentration or, equivalently, its resultant negative reactivity effect behaves as

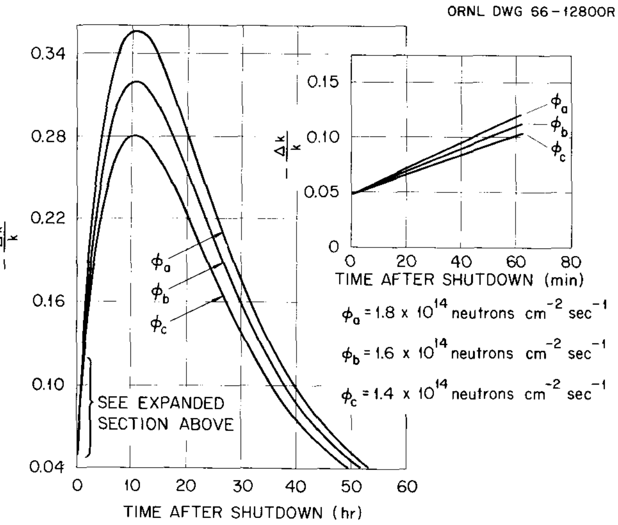


Fig. 7.24. Poisoning Due to Xenon Buildup After Shutdown.

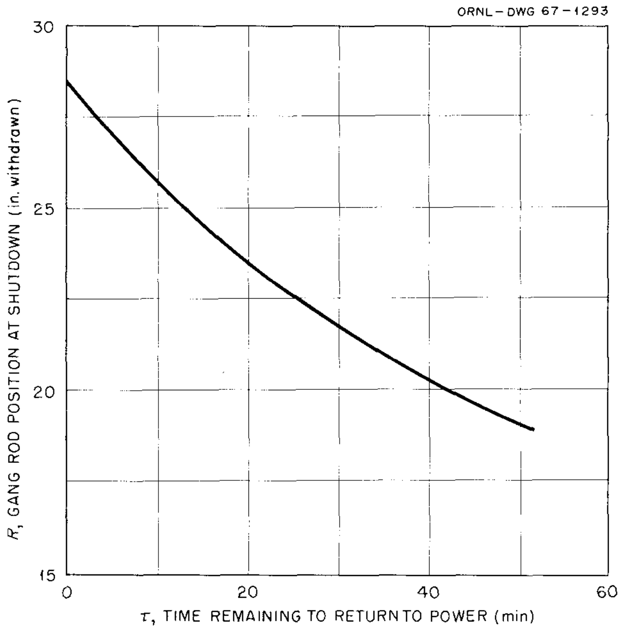


Fig. 7.25. Time Required to Return to Power After Accidental Shutdown vs Rod Position at Shutdown.

shown in Fig. 7.24. As a result of this xenon buildup, the reactor must be restarted according to the schedule indicated by Fig. 7.25, or the refueling operation described previously must be performed. During routine fuel reloadings, most of the fuel elements are replaced by new elements or partially burned elements in which the xenon has been decaying for one or more fuel cycles. Therefore, the xenon poisoning at the beginning of a fuel cycle is of little or no concern.

Samarium poisoning, whose effect is considerably smaller in magnitude than the effect of xenon poisoning, is still large enough to be significant. The negative reactivity contribution at its equilibrium value during operation is approximately $-0.0097\ \Delta k/k$. The major concern with respect to the samarium concentration arises from the fact that, because ^{149}Sm is for all practical purposes a stable isotope, its concentration increases to a value after the reactor is shut down that is higher than its steady-state operating value, and it does not decay off after reaching its peak value, as in the case of the xenon poisoning. Thus, partially burned elements which are stored for xenon decay and then used again in another fuel loading contain this larger concentration of ^{149}Sm. After reactor startup, the samarium is burned until its concentration reaches a new equilibrium value. This results in an initial addition of reactivity which partially compensates for the buildup of xenon.[6]

The calculation of fuel burnup (Appendix C) during a cycle is performed routinely for each fuel element,[7] and detailed records are kept on each of the elements on hand. These records are used in determining elements to be used in subsequent core loadings and allow accurate calculations of the reactivities of particular cores.

Detailed investigations[8] have been performed to determine the fuel burnup with respect to longitudinal position in the fuel elements and the fuel followers of control rods. In general, the results of these investigations indicate that the burnup is not symmetric about the vertical center line of the reactor but is distorted slightly, the maximum burnup occurring 2 to 4 in. below the center line. Essentially the same results were obtained for the fuel followers of the shim rods. The distortion of the burnup should be expected, though, since the neutron flux is also distorted into that same general configuration.

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[6] J. A. Cox, *ORR Operations for Period April 1961 to April 1962*, ORNL-TM-351, pp. 78–80 (Oct. 16, 1962).

[7] T. P. Hamrick and H. F. Stringfield, *Proceedings of the AEC-Contractor Nuclear Fuels Management Conference, Lawrence Radiation Laboratory, October 1966*, UCID-15020.

[8] J. A. Cox, *ORR Operations for Period April 1959 to April 1960*, ORNL-CF-60-9-2, pp. 60–78 (Sept. 30, 1960).

7.3.4 Temperature, Void, and Fuel Coefficients of Reactivity

Temperature, void, and fuel coefficients have been determined experimentally in the ORR core.[9–12]

A plot of core reactivities vs core temperature for the clean fuel loadings shown previously in Figs. 7.6 through 7.8 are shown in Fig. 7.26. As may be noted, the slopes of these curves are negative for all values of the temperature given, so the temperature coefficients are negative. It may also be noted that there is little difference in the slopes of the curves, so it may be assumed that changing the core loading and configuration has only a small effect on the temperature coefficient. Measurements on a recently operated core (including fission products) indicate that the temperature coefficient may be expressed as

$$\alpha_T = -0.006554 - 0.0000121T ,$$

where T is in degrees Fahrenheit and α_T has units $\Delta k/k$ per degree Fahrenheit. An average value of the temperature coefficient over the range 70 to 110°F is 7.5×10^{-5} $\Delta k/k$ per degree. The change in reactivity associated with an overall isothermal increase in core temperature from 70 to 125°F amounts to about −0.41% $\Delta k/k$.

An experimental measurement of the moderator-coolant void coefficient was made in position *B*-3 of the fuel region. The results of that measurement yielded a value of about 1.5×10^{-6} $\Delta k/k$ per cubic centimeter.

Fuel coefficients were determined experimentally for the original clean core conditions.[9] Since the reactivity effect of a fuel addition varies so much from one core position to another, no one fuel coefficient measurement is valid for the core as a whole. Instead, the reactivity effect must be measured for each location and then expressed as some convenient function of the local flux.

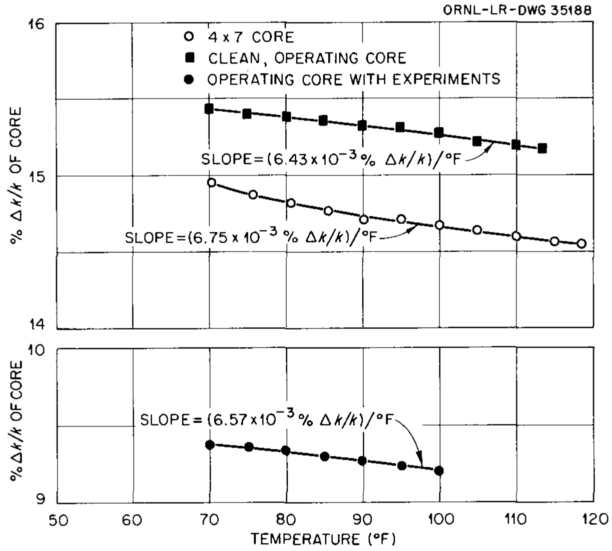

Fig. 7.26. Core Reactivity Changes vs Core Temperature.

The change in reactivities due to fuel additions into the first operating core and the revised operating core configurations were measured with the experiments in place. Agreement of the measurements on the two cores was good. The results of these measurements led to the adoption of the empirical relation:

$$(\Delta k/k) \approx (\Delta m_i/m)\,(\bar{\Phi}_i/\Phi_{max}) .$$

The Δm_i is the weight of fuel added in core position i, where the average thermal neutron flux was $\bar{\Phi}_i$, and m is the total weight of fuel in the core before the addition. The Φ_{max} is the maximum thermal flux in the core. Other measurements on cores used more recently indicate that the following, somewhat more convenient relationship also applies:

$$(\Delta k/k) = 0.454\,(\Delta m_i/m)(\bar{\Phi}_i/\bar{\Phi}_c) ,$$

where $\bar{\Phi}_c$ is the average thermal flux in the core.

These relationships are limited to small values of Δm_i; that is, Δm_i should be a small fraction of the total core weight. Therefore, for small Δm_i the change in reactivity for any change in fuel loading can be estimated, if the amount of fuel present just prior to the change is known.

[9] C. D. Cagle and R. A. Costner, Jr., *Initial Post-Neutron Measurements in the ORR*, ORNL-2559 (May 28, 1959).

[10] W. H. Tabor and S. S. Hurt III, *Oak Ridge Research Reactor Quarterly Report: April, May and June of 1965*, ORNL-TM-1920, pp. 30–34 (Oct. 6, 1965).

[11] T. P. Hamrick, "Experimental Evaluation of Errors Arising from the Burnup Calculations Used at the ORNL Research Reactor," *Trans. Am. Nucl. Soc.* 8(suppl), 25–26 (1965).

[12] T. P. Hamrick to F. T. Binford, "Measurement of Temperature Coefficient in the ORR," unpublished correspondence.

7.3.5 Reactivity Associated with Experimental Rigs and Experimental Facilities

The reactivity effects of experimental rigs and experimental facilities of the ORR are of considerable importance because replacement of fuel or beryllium reflector elements with absorber- or fuel-containing experimental facilities may have a significant effect upon the operating characteristics of the reactor. Naturally, the reactivity effect produced by an experimental rig depends upon the configuration of the core and the rig's relative position, but it also depends to a large degree on the total mass of fuel in the core. A number of experimental rigs which are representative of those normally inserted in the ORR and their accompanying effects on the reactivity have been listed in Table 7.3.[13] No detailed specifications of the experimental rigs are given, but a brief description of their general type and configuration is listed in the table.

The beam holes and large north and south facilities may be operated as voids or filled with water. When any of these voids are filled with water or some other material, the net neutron leakage from the core is altered, and, consequently, a change in reactivity occurs. Reactivity measurements of these effects were made shortly after the initial critical run on the 4×7 core (Fig. 7.5) and the clean first operating core (Fig. 7.6). The reactivity changes caused by emptying the various regions were measured and are presented in Table 7.4. For the 4×7 core, assuming that the effect of draining HS is the same as that for HN and that of draining HB-4 is the same as that for HB-3, the total worth of draining all the facilities is -1.14% $\Delta k/k$. For the operating core, the same assumptions give a total worth of -1.99% $\Delta k/k$ for draining all the facilities. Conversely, filling the beam holes should result in an addition of $+1.14\%$ or $+1.99\%$ $\Delta k/k$ for the respective cores.

7.3.6 Nuclear Characteristics of Shim Rods

The control system used in the ORR was designed primarily to effectively control reactivity without causing unnecessarily large perturbations in the power distribution and without causing undesirable reactor shutdowns due to control system malfunctions. The control of the reactor is effected by the vertical positioning of six removable shim rods within the reactor (Fig. 7.1). Two of the shim rods (Nos. 1 and 2) may have either aluminum or fuel followers, while the other four (3 through 6) have fuel followers. Each of the six shim rods has its own drive rod and release mechanism, and they operate completely independently of each other. In the event of an emergency, they are released separately, thereby providing multiplicity of control. Releasing any one of the shim rods will shut the reactor down by itself.

During normal operation, Nos. 1 and 2 are fully withdrawn, so that only the follower portion of the control rod is located in the core region. Generally, these rods are not used for shim and/or regulation but instead are used only to provide an extra shutdown margin. The remaining four rods are used for shim operations and are maintained in as nearly equally withdrawn positions as practical. Number 6 is normally used as the servo-controlled regulating rod.

In order to maintain the vertical power distribution within acceptable limits and to decrease the possibility of flux peaking in water gaps in positions from which control rods would be withdrawn, either fuel or aluminum followers were added to the lower ends of the control rods.

The poison sections of the control rods are made of aluminum-clad cadmium sheet which is formed into square-cross-section cans that fit into the regular lattice positions of the core. The fuel followers are very similar to a regular fuel element; the main difference is that only 14 fuel plates are used in each follower. The total fuel content in a new shim follower is 154 g of ^{235}U contained in aluminum-clad uranium-aluminum alloy fuel sections. Uranium used in the 19.5% U-Al alloy is enriched to 93 wt % ^{235}U. Detailed specifications of the composition of the shim rods are given in Tables 5.4, 5.5, and 5.6. The aluminum followers are simply solid blocks of 1100 aluminum with $\frac{1}{2}$-in.-diam cooling channels drilled along their vertical axes. The horizontal cross sections of the followers have the same overall dimensions as the fuel-follower parts of the other shim rods.

Differential and integral reactivity worths for the control rods have been determined experimentally for a number of core loadings. Extensive rod calibration experiments were performed on the original

[13] W. H. Tabor to F. T. Binford, "Current Operating Experiments in the ORR," unpublished correspondence.

Table 7.3. Reactivity Effects of Experimental Rigs[a]

Core Position or Facility	Nature of Experimental Rig	Reactivity Worth (% $\Delta k/k$)
A-2	Refractory fuel element in capsule	−0.06
A-3	^{235}U fuel ring for iodine production	Not measured
A-7	Utility tray with various small capsules	−0.96[b]
A-9	$(ThU)C_2$ particles in capsule	Insignificant
B-8	High-temperature Be irradiation in large capsule	−0.14
B-9	UO_2 fuel sample (0.32 g ^{235}U) in capsule	Insignificant
C-1	UO_2 fuel sample (0.10 g ^{235}U) in capsule	−0.35[b]
D-8	Various encapsulated materials in hydraulic tube[c]	$\lvert\Delta k/k\rvert \leqq 0.27\%$
F-3	Six UO_2 fuel capsules ($\sim$3.6 g ^{235}U per capsule)	−0.27[d]
F-5	Various encapsulated isotopes for irradiation	Not measured
F-8	Various encapsulated materials in hydraulic tube[c]	$\lvert\Delta k/k\rvert \leqq 0.27\%$
F-9	UO_2 fuel for meltdown (encapsulated)	+0.01
	UO_2 high-burnup capsule	Insignificant
P-5	Creep test samples (in large capsule)	−0.12

[a]Effective as of January 1, 1967.

[b]The reactivity worth measured is the net effect produced by replacement of a beryllium reflector element by the experiment under consideration.

[c]Materials with a large thermal neutron absorption cross section are limited to an amount equivalent to an effective worth of 5 cm^2 of projected area of cadmium. Rapid removal of a sample with an effective worth of 5 cm^2 of projected area of cadmium will produce a reactor period of $\sim$15 sec, which corresponds to a reactivity of $\sim$0.27% $\Delta k/k$.

[d]This value was determined based on the measurement of −0.045% $\Delta k/k$ for a single capsule.

core loadings during the post-neutron measurements.[14] The calibrations were performed by introducing a neutron absorber into the core and determining the resultant reactivity worth of the movement of the rods required to keep the reactor critical. It should be emphasized that these rod-worth measurements were made with approximately a uniform fuel distribution which is not that which normally exists in the ORR. More recent calibrations[15,16] have been made on a typical operating core by using the conventional period method. Results of the tests on the fuel-follower shim rods are shown in Fig. 7.27 for the individual rods and in Fig. 7.28 for all four of them ganged. The two aluminum-cadmium shim rods were also calibrated individually and together by the same method. Results of those tests are shown in Figs. 7.29 through 7.31. Total reactivity worths of the rods

Table 7.4. Summary of Void Worths

Condition	Void Worths (% $\Delta k/k$)	
	4 × 7 Core	Operating Core
HB-1 drained	−0.035	−0.038
HB-1 and HB-2 drained	−0.049	−0.042
HB-1, HB-2, and HB-3 drained	−0.074	−0.113
HB-2 and HB-3 drained	−0.054	−0.080
HB-3 drained	−0.047	−0.065
HB-1, HB-2, HB-3, HB-5, and HB-6 drained	−0.127	−0.201
HN drained	−0.470	−0.863

[14]C. D. Cagle and R. A. Costner, Jr., *Initial Post-Neutron Measurements in the ORR*, ORNL-2559 (May 28, 1959).

[15]T. P. Hamrick to F. T. Binford, "Calibration of ORR Shim Rods," unpublished correspondence.

[16]T. P. Hamrick to F. T. Binford, "Calibration of No. 1 and 2 Shim Rods – ORR," unpublished correspondence.

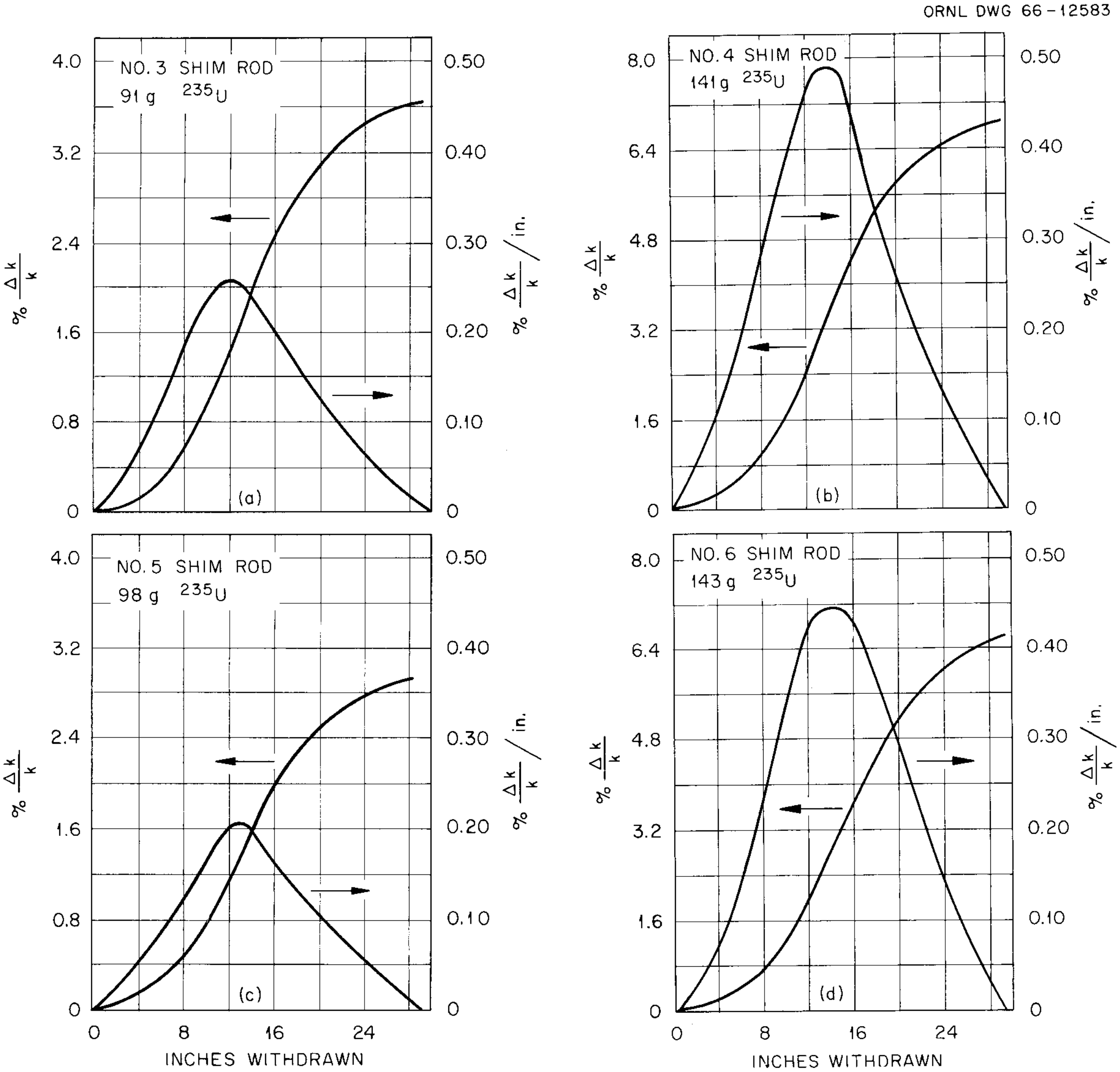

Fig. 7.27. Numbers 3, 4, 5, and 6 Shim-Rod Calibration in the ORR.

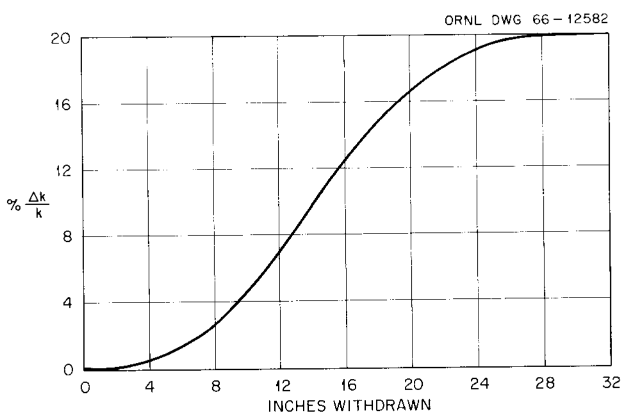

Fig. 7.28. Gang-Rod Calibration.

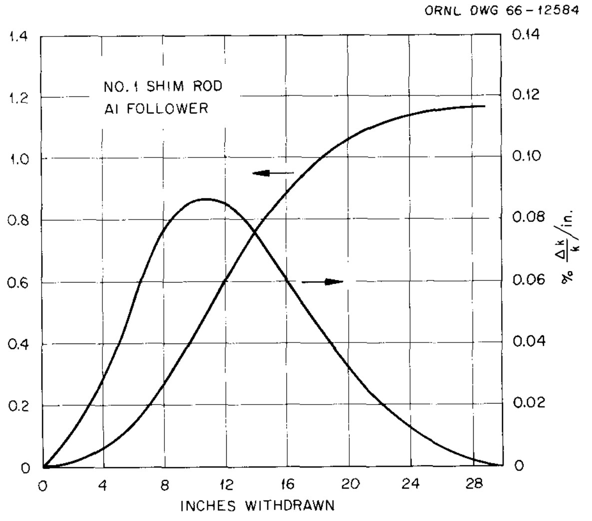

Fig. 7.29. Number 1 Shim-Rod Calibration in the ORR.

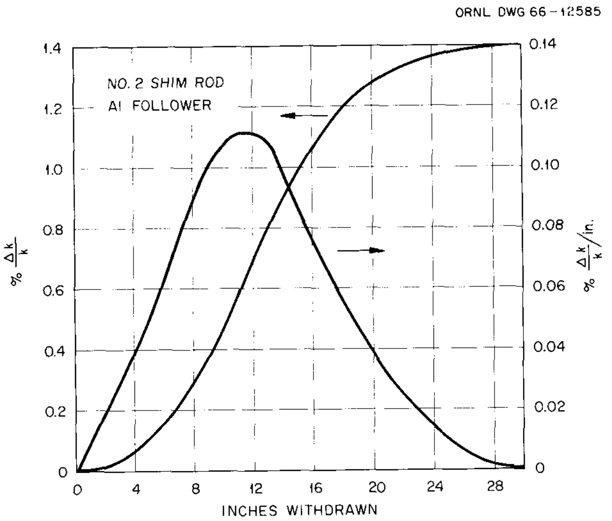

Fig. 7.30. Number 2 Shim-Rod Calibration in the ORR.

are indicated in Table 7.5. It must be emphasized that these rod calibrations are strictly valid only for this particular core loading, but, for most normal loadings, these values will be reasonably good.

7.3.7 Reactivity Accountability

Reactivities associated with a typical ORR core are given in Table 7.6. Since some of the values change from one core loading to another, these values should be considered only as representative estimates.

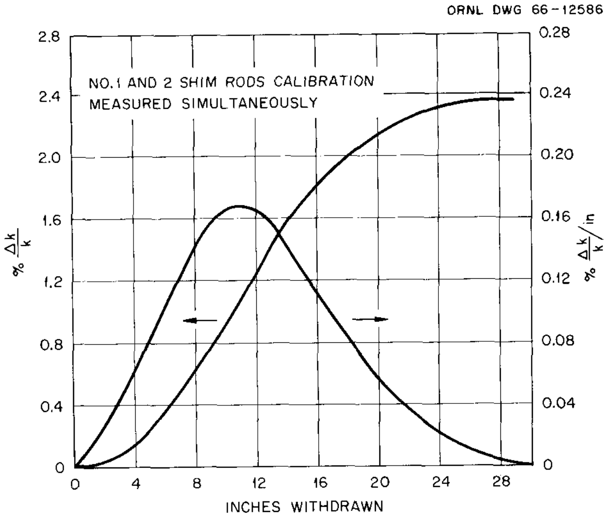

Fig. 7.31. Combined Worth of Nos. 1 and 2 Shim Rods in the ORR.

Table 7.5. Shim-Rod Reactivity Worth

Rod Number	Reactivity Worth (% $\Delta k/k$)
1	1.16
2	1.40
1 and 2	2.36
3	3.63
4	6.84
5	2.95
6	6.49

7.4 Fuel Element Design and Analysis

The design of the ORR fuel elements has evolved from the initial design, which was essentially the same as that for the MTR fuel elements, to the present design, which is a considerable improvement over the first. Of necessity, the detailed design was influenced by the primary purposes of the reactor and by the characteristics desired for the fuel elements themselves.

7.4.1 Fuel Element Design

Consideration of several types of fuel elements with respect to technological and economic considerations led to the choice of the approximately

Table 7.6. Summary of Reactivities for a Typical ORR Core

Parameter	Reactivity Worth at Equilibrium (% $\Delta k/k$)
Fuel worth (zero power at 70°F)	14
Temperature (isothermal increase from 70 to 125°F)	0.41
Temperature (zero power at 70°F to 30 Mw)	−0.43
^{135}Xe	−4.70
^{149}Sm	−0.97
Fission products (other than ^{135}Xe and ^{149}Sm)	−0.16
In-reactor experiments (typical value)	2
Beam-hole flooding	+0.27
Engineering facility flooding	+1.73
Control-rod worth	26
Shutdown margin	13

square-cross-section rectangular plate-type elements using aluminum-clad uranium-aluminum alloy fuel plates. The technology of the plate-type elements chosen was more advanced than it was for other types of fuel elements. The development and use of that type of element in the MTR contributed enormously to the knowledge and confidence in the use of the plate-type element, and, since the majority of the developmental work was complete, the economic aspects of that design were also very attractive.

The thickness of the fuel-alloy core in each fuel plate is the same across the width of the fuel plate. No attempt has been made to vary the fuel distribution either longitudinally or transversely within the elements, because the benefits that might be realized from that practice would be far offset by the additional cost of fabrication of the elements.

The fuel plates were specified to have a $5\frac{1}{2}$-in. radius of curvature, because curved plates have desirable characteristics with respect to thermal and hydraulic loads and stresses. With the exception of the thicker-clad shim-rod-follower plates and outer fuel element plates, all of the fuel plates are the same thickness and have the same fuel loading.

7.4.2 Design Criteria and Analyses

Some of the performance demands imposed upon fuel elements were: (1) coolant velocities on the order of 30 fps, (2) coolant channels and gaps between core components as small as 0.100 in. or less, (3) fuel plates as thin as 0.050 in., (4) fuel-plate surface temperatures up to 210°F, and (5) heat fluxes up to 6×10^5 Btu hr^{-1} ft^{-2}. This group of characteristics imposed upon the design introduced several problems concerning metallurgical, mechanical, and hydraulic considerations. In order to determine many of the characteristics of the elements when used under the above-mentioned circumstances, experimental programs were initiated to obtain the desired information. Among the several studies performed, some of the more important general programs investigated were: (1) corrosion in the ORR core-cooling system, (2) operating characteristics of the ORR cooling system, (3) heat transfer and hydraulic characteristics of the ORR fuel elements and core, and (4) gamma heating of reactor components. Reports of the investigations and analyses have been presented in Appendix E and will not be reproduced here. As a result of those studies and others, considerably improved core components are now being used.

7.4.3 Mechanical and Hydraulic Analyses

The curved fuel plates used in the ORR are subjected to two main kinds of loading. One type is that which results from thermally induced differential transverse and longitudinal expansions of fuel plates, and the other is that which results from lateral pressure differentials existing across the fuel plates. In general, elements with curved fuel plates present more complex design and analysis problems than ones with flat fuel plates would. However, the curved plates are structurally stronger than flat ones; hence, the curved-plate-type elements were chosen for the ORR.

The temperature increase of the fuel elements in increasing the power from zero to 30 Mw results in thermal expansion of the fuel plates both in the longitudinal and transverse direction. Transverse expansion is very small and results only in a slight increase of the arc of the plate or, alternatively, a decrease in the radius of curvature of the arc. In some cases, this expansion might have a tendency to decrease the thickness of a coolant channel, but, since all the plates in the element expand about the same amount, the thicknesses of coolant channels remain nearly the same. The maximum thermal expansion of the plates' thicknesses is even less important than the transverse expansion. The longitudinal expansion is the most significant of the three types; however, even this expansion is of little concern. Since the two side plates of the fuel elements contain fuel, no important temperature differences exist between the side plates and the inner fuel plates. Therefore, the difference in expansion is small, so the inner plates do not have to withstand large compression loads. Furthermore, since the fuel elements themselves are not held rigidly in place with respect to their vertical position, the element may expand freely without constraint.

Hydraulic loads exist because of small variations in plate dimensions, which in turn cause differences in coolant channel widths. These loads would exist on the plates whether the fuel plates were curved or not; therefore, this type of loading is not unique to the ORR's curved-plate-type elements. The net result of a small differential pressure across a fuel plate would be for the plate to be strained so that the radius of curvature of the plate would increase or decrease depending on whether the net pressure was greater on the convex or concave side of the plate. This effect then would increase the flow area in one coolant channel while decreasing it in the adjacent one. The critical velocity for buckling is higher for the curved fuel plate than it is for the flat plate, because the curved plate simply is structurally stronger both with respect to bending and torsional loads.

7.5 Hydraulic and Thermal Characteristics of the Core

The hydraulic and thermal characteristics of the ORR core were examined in considerable detail before extensive high-power operation was begun. Calculations[17] of velocities and flow rates were made for many specific locations in the reactor core as well as for the composite arrangement. The water velocities through the fuel sections that were required in order to attain particular fuel element surface temperatures were calculated on the basis of a maximum heat flux for a 28-element core of 6×10^5 Btu ft^{-2} hr^{-1} and an inlet coolant temperature of 120°F. Results of those calculations for the core in Fig. 7.32 are shown in Fig. 7.33. In order to limit the maximum surface temperature to $\sim$210°F (30°F below the saturation temperature of H_2O at operating pressure), a velocity of 32 fps through the fuel with a total flow of 18,000 gpm was required. The results of flow measurements[18] through core components are shown in Figs. 7.34 through 7.40. The distribution of flow through the fuel elements and in the channels between curved sides of the elements is given in Table 7.7. The water velocity in the outermost channels of fuel elements is as much as 17% less than that in the central channels. Therefore, so far as the fuel elements are concerned, the power of the reactor will be limited by the temperature of the outer fuel plates in regions of high neutron flux.

The coolant flow through the reactor is a function of the pressure differential across the reactor tank. Figure 7.41 shows this differential value as a function of coolant flow.

Calculations[19] of several plate temperature profiles were made for certain special conditions, and the results of those calculations are shown in Fig. 7.42. For the cases shown, it was assumed that the reactor had recently been started up. The experimental data used were taken for a core that had recently been started up, and the element used as a basis for these calculations was a new 240-g element. It was further assumed that the fuel plate surface temperature did not reach the boiling point of the coolant water. These calculations appear to be somewhat conservative, but they indicate the longitudinal temperature distribution along a fuel plate in the startup condition which has associated with it the highest localized heat fluxes.

[17]F. T. Binford, *Hydraulic and Thermal Characteristics of the ORR Core*, ORNL-CF-55-5-157 (May 27, 1955).

[18]F. T. Binford, *Preliminary Report on the Results of the Oak Ridge Research Reactor Hydraulic Test*, ORNL-CF-58-2-11 (Feb. 17, 1958).

[19]C. C. Webster, "Analysis of the Fuel Plate Surface Temperatures of an ORR Fuel Element," to be published as an ORNL report.

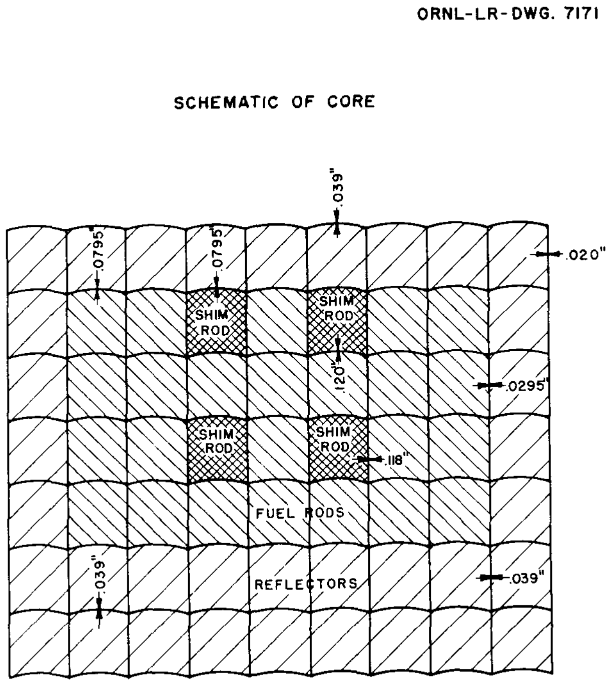

Fig. 7.32. ORR Hydraulic Test Core.

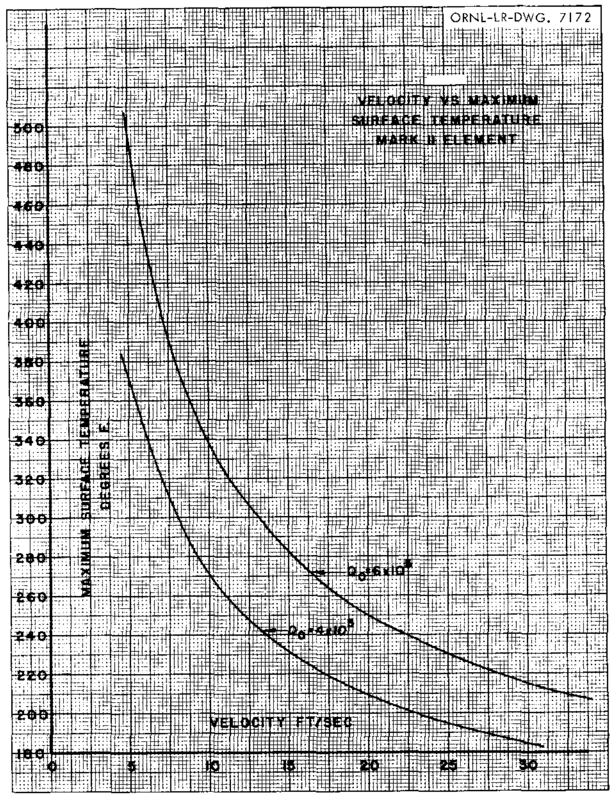

Fig. 7.33. Velocity vs Maximum Surface Temperature in ORR Fuel Element.

A hot-spot, hot-channel analysis was performed[20] on the ORR fuel elements. Such items as variation in fuel-plate density, variations in coolant gaps, presence of occlusions, and other similar design and construction irregularities were studied and incorporated into the analysis.

The requirements for afterheat removal for 30-Mw operation of the ORR were investigated[21] using the criterion that boiling would not occur at any point in the reactor. This criterion, then, limited the surface temperature at the hottest spot to a value below the temperature of incipient nucleate boiling. The results of the investigation indicated that a flow of 1000 gpm is desired after shutdown, but in most cases a flow of 500 gpm will minimize the probability of a meltdown. Figure 7.43 is a graph of the fuel-plate surface temperature for various shutdown flow rates. The curve entitled "hottest channel" takes into account the decrease in velocity which is present in the outer fuel plates.

In summary, the ORR fuel elements and core plates perform quite adequately with respect to their heat transfer and hydraulic characteristics. For operation at 30 Mw and a coolant flow of about 18,000 gpm, the fuel element temperatures, mechanical and hydraulic stresses, and the coolant temperature are well within specified limits during normal operation.

[20] J. F. Wett, Jr., *Some Heat Transfer Characteristics of ORR Fuel Elements*, ORNL-CF-61-1-49 (Jan. 19, 1961).

[21] J. F. Wett, Jr., *Requirements for Afterheat Removal for 30-Mw Operation of the ORR*, ORNL-CF-60-6-13 (June 7, 1960).

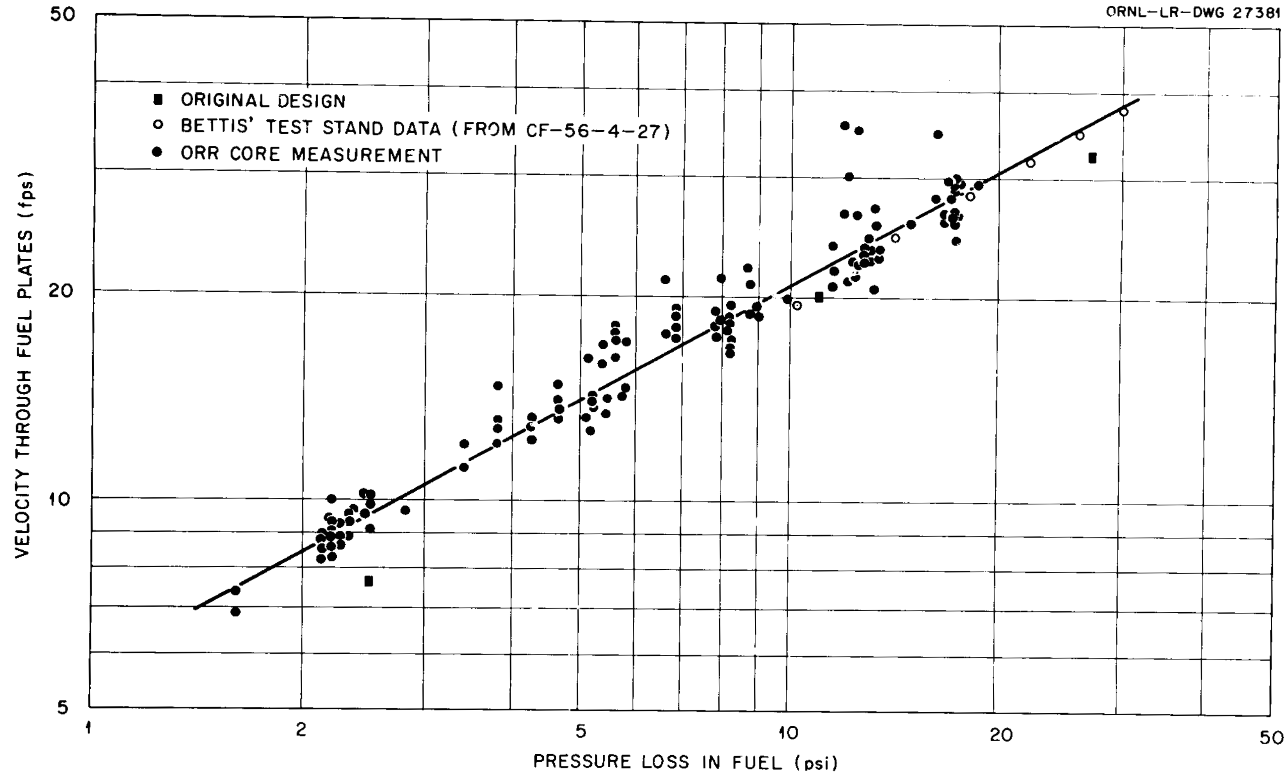

Fig. 7.34. Velocity Through ORR Fuel Element Cooling Channels vs Pressure Drop Across the Core.

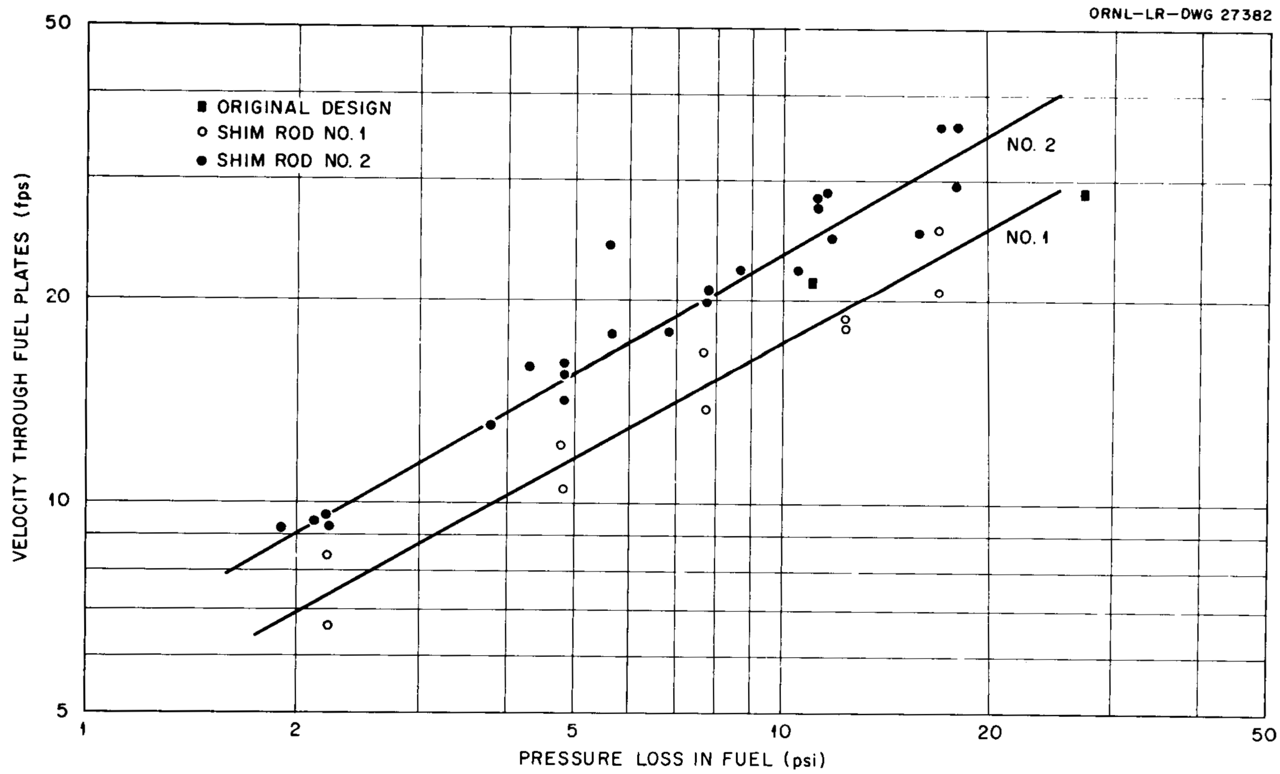

Fig. 7.35. Velocity Through ORR Fuel Element Cooling Channels vs Pressure Drop Across the Core.

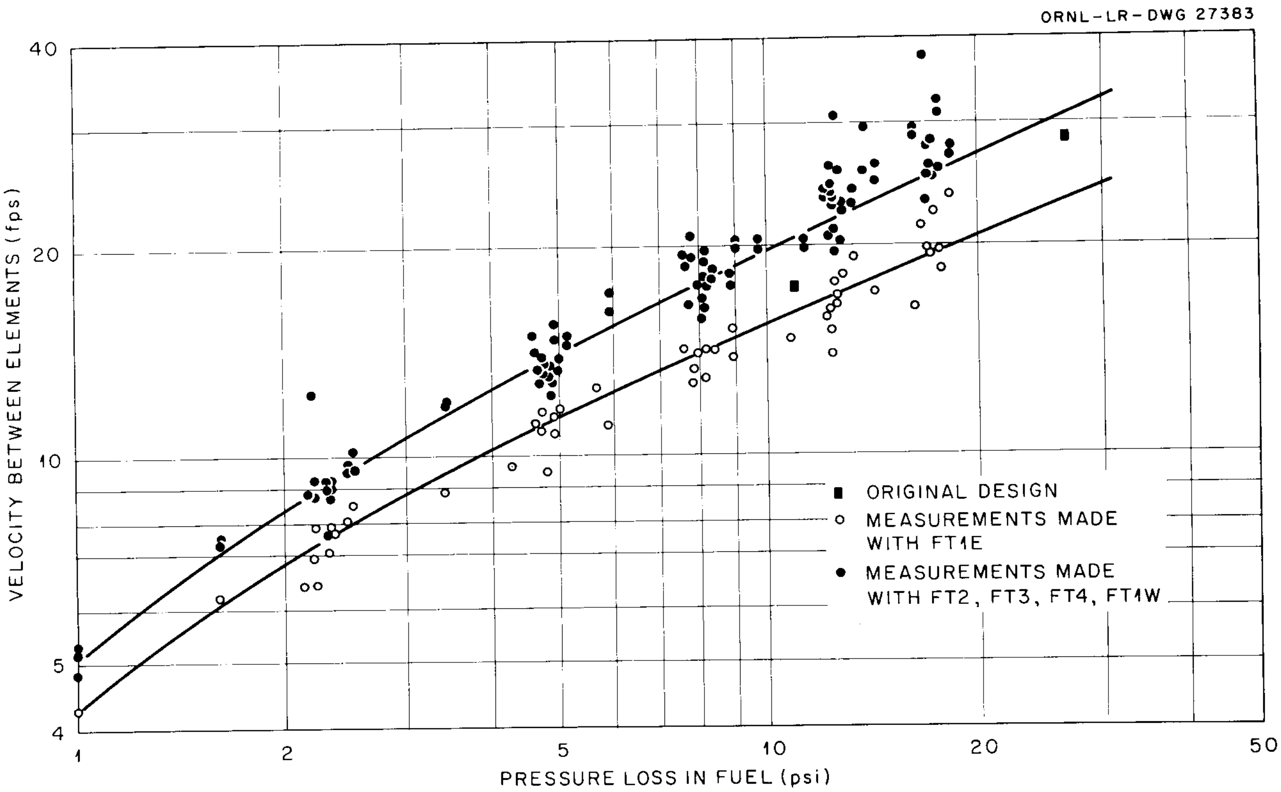

Fig. 7.36. Velocity Through ORR Fuel Element Cooling Channels vs Pressure Drop Across the Core.

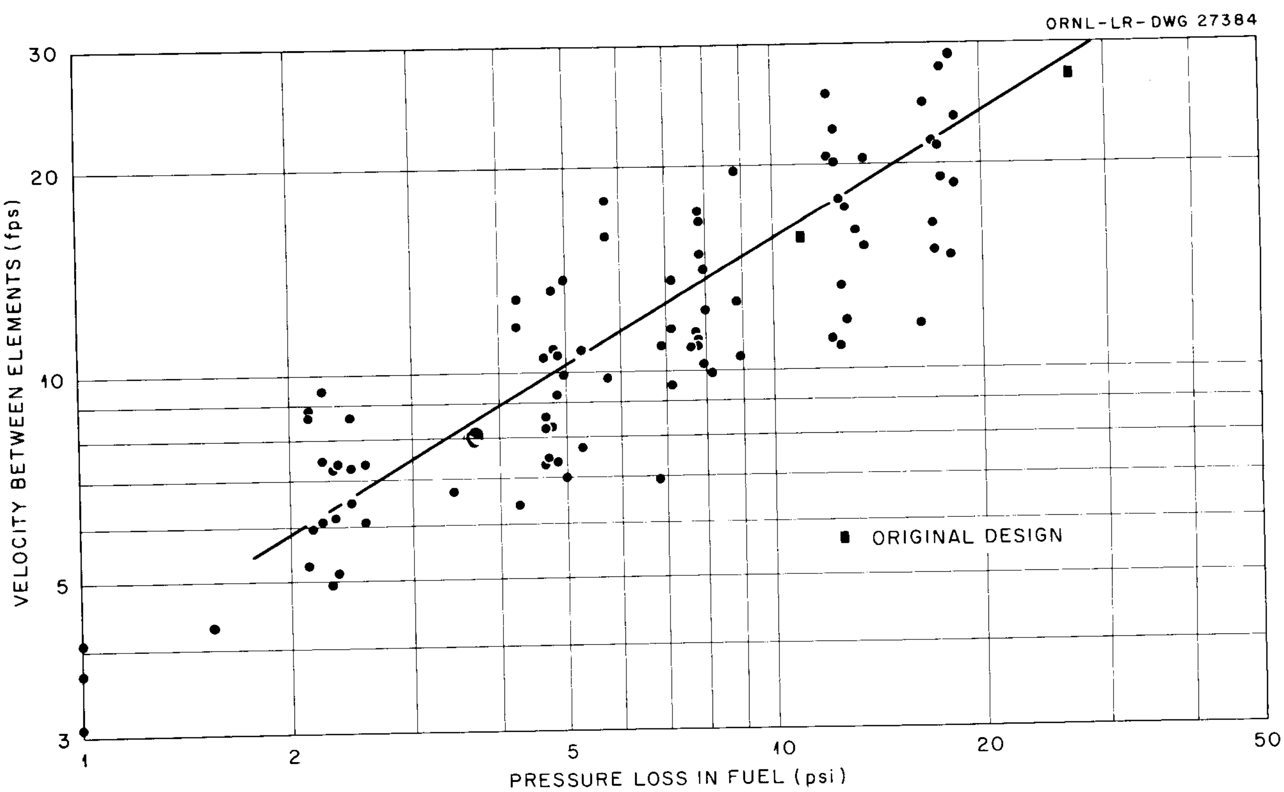

Fig. 7.37. Velocity Through ORR Fuel Element Cooling Channels vs Pressure Drop Across the Core.

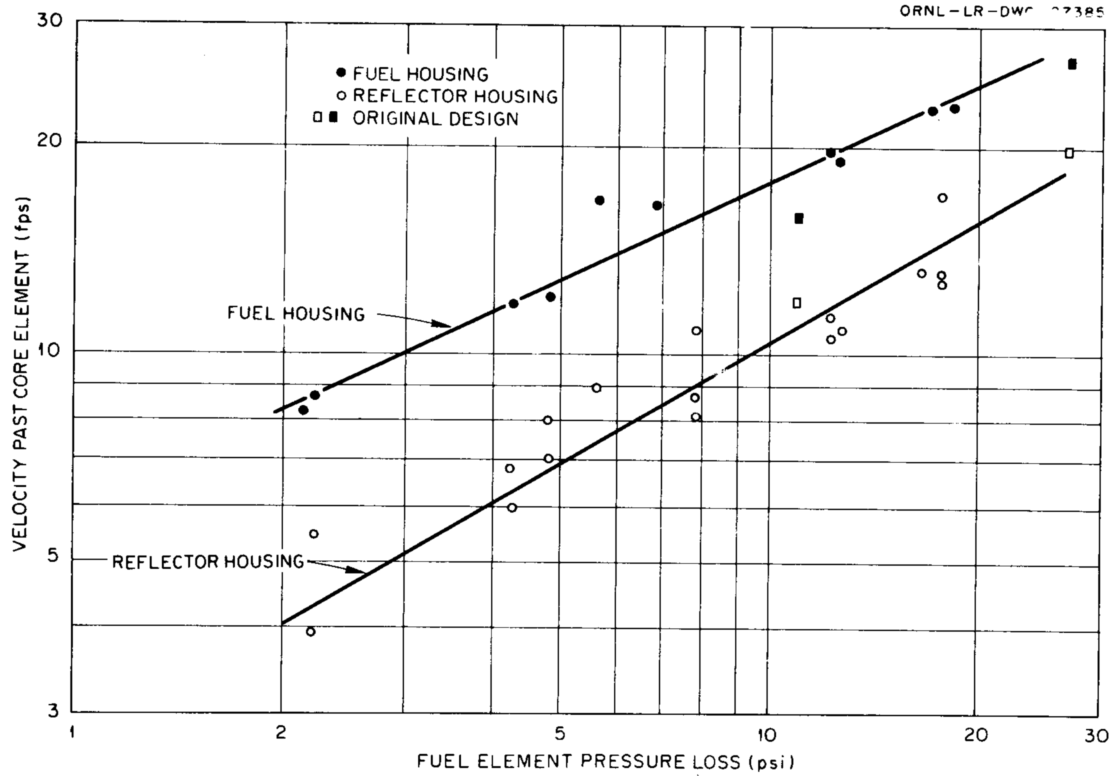

Fig. 7.38. Velocity Through ORR Fuel Element Cooling Channels vs Pressure Drop Across the Core.

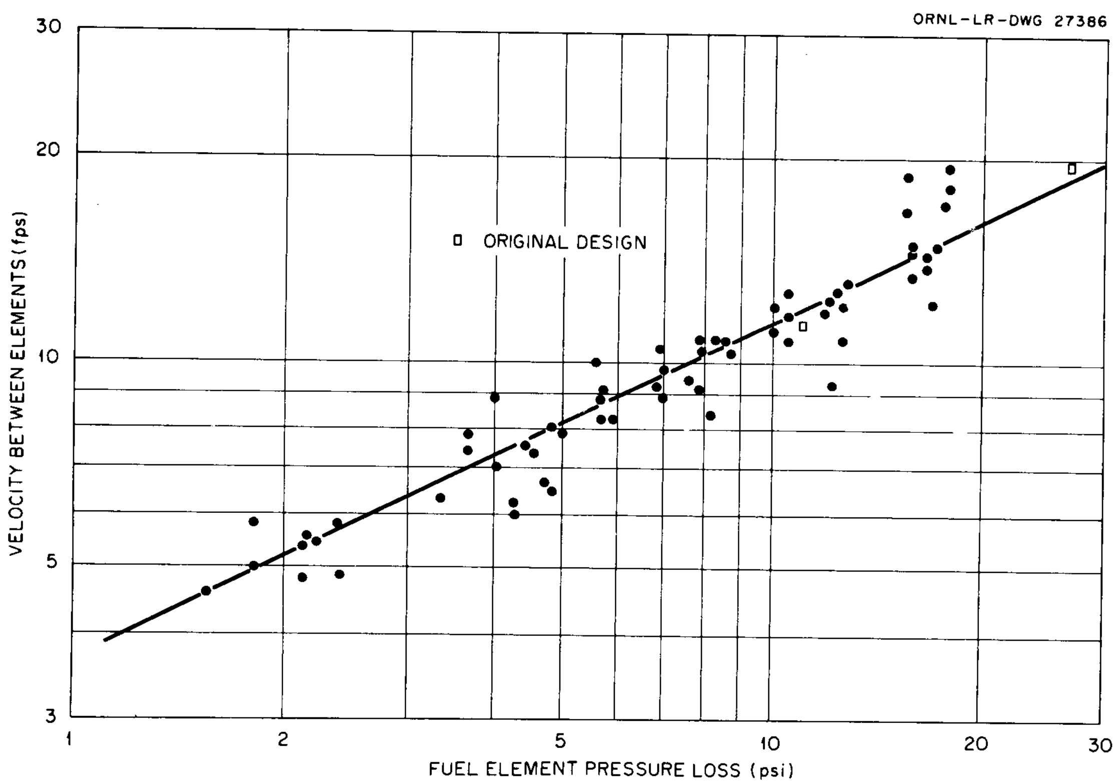

Fig. 7.39. Velocity Through ORR Fuel Element Cooling Channels vs Pressure Drop Across the Core.

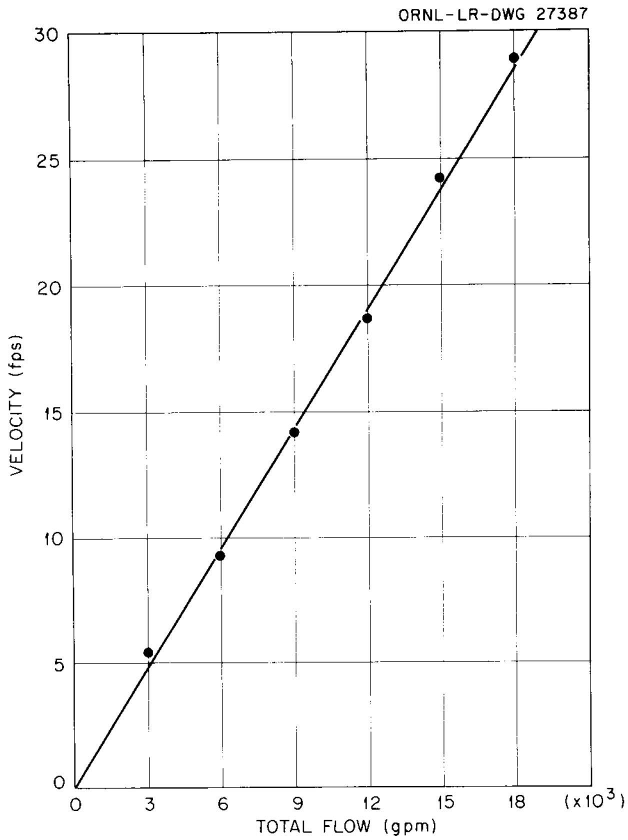

Fig. 7.40. Total Flow Through ORR Core vs Coolant Velocity.

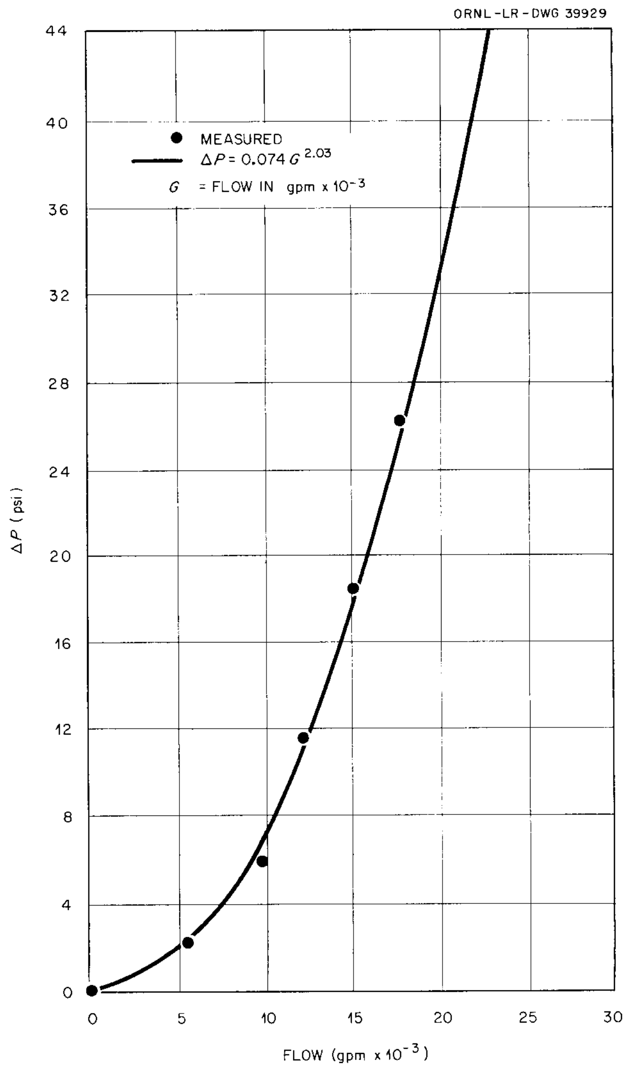

Fig. 7.41. Pressure Loss Across the ORR Tank vs Total Flow.

Table 7.7. Distribution of Flow Through ORR Core (at 120°F)

	Flows (gpm) at Pressure Losses of			
	4.1 psi	8.9 psi	13.6 psi	19.2 psi
Through 26 fuel elements	5643	8,733	10,972	13,435
Through 4 shim rods	703	1,085	1,425	1,759
Through 15 reflector annuli	210	315	390	480
Through 12 reflector holes	23	34	43	54
Between 21 fuel–fuel curved sides	301	462	533	628
Between 25 fuel–fuel flat sides*	68	98	122	144
Between 3 fuel–housing curved sides	26	38	46	54
Between 4 fuel–housing flat sides*	6	10	12	15
Between 15 fuel–reflector curved sides	95	152	198	240
Between 6 fuel–reflector flat sides*	10	15	19	22
Between 18 reflector–reflector curved sides	100	155	179	350
Between 25 reflector–reflector flat sides*	23	33	41	49
Between 15 reflector–housing curved sides	34	53	69	84
Between 10 reflector–housing flat sides*	9	13	17	20
Rifle-drilled holes	174	245	303	359
Large facility annuli	221	317	368	469
Exterior ion chamber shield	22	32	38	46
Interior ion chamber shield	8	13	15	21
Beam holes HB1–HB6	224	288	407	513
D_2O system	138	204	254	312
Total	8038	12,295	15,451	19,054
Motored flow	8000	12,200	15,300	18,700

*Calculated values from ORNL CF-55-5-157; these values represent at most about 2½% total flow.

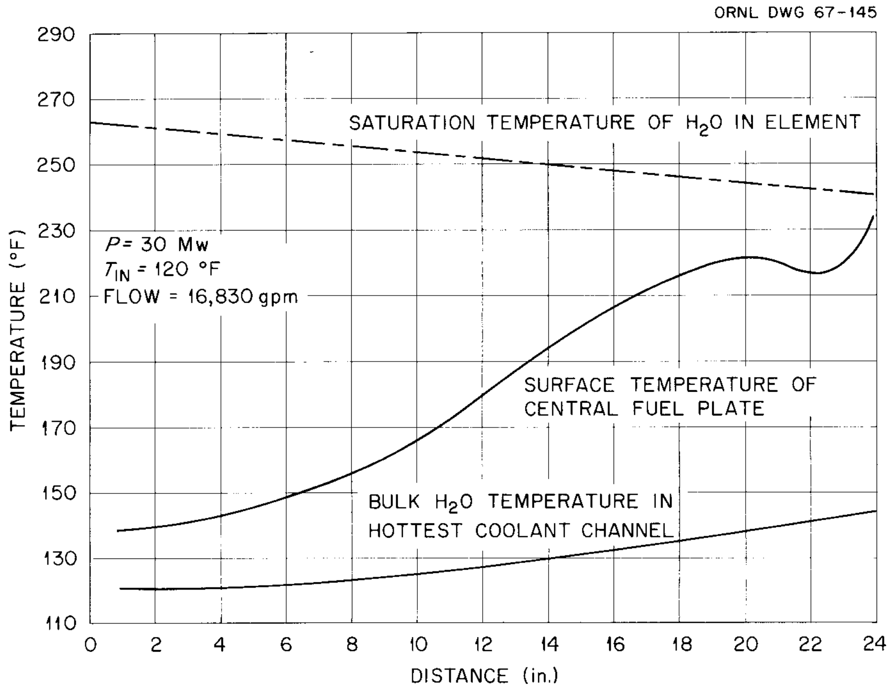

Fig. 7.42. Temperature Profiles of ORR Fuel Elements.

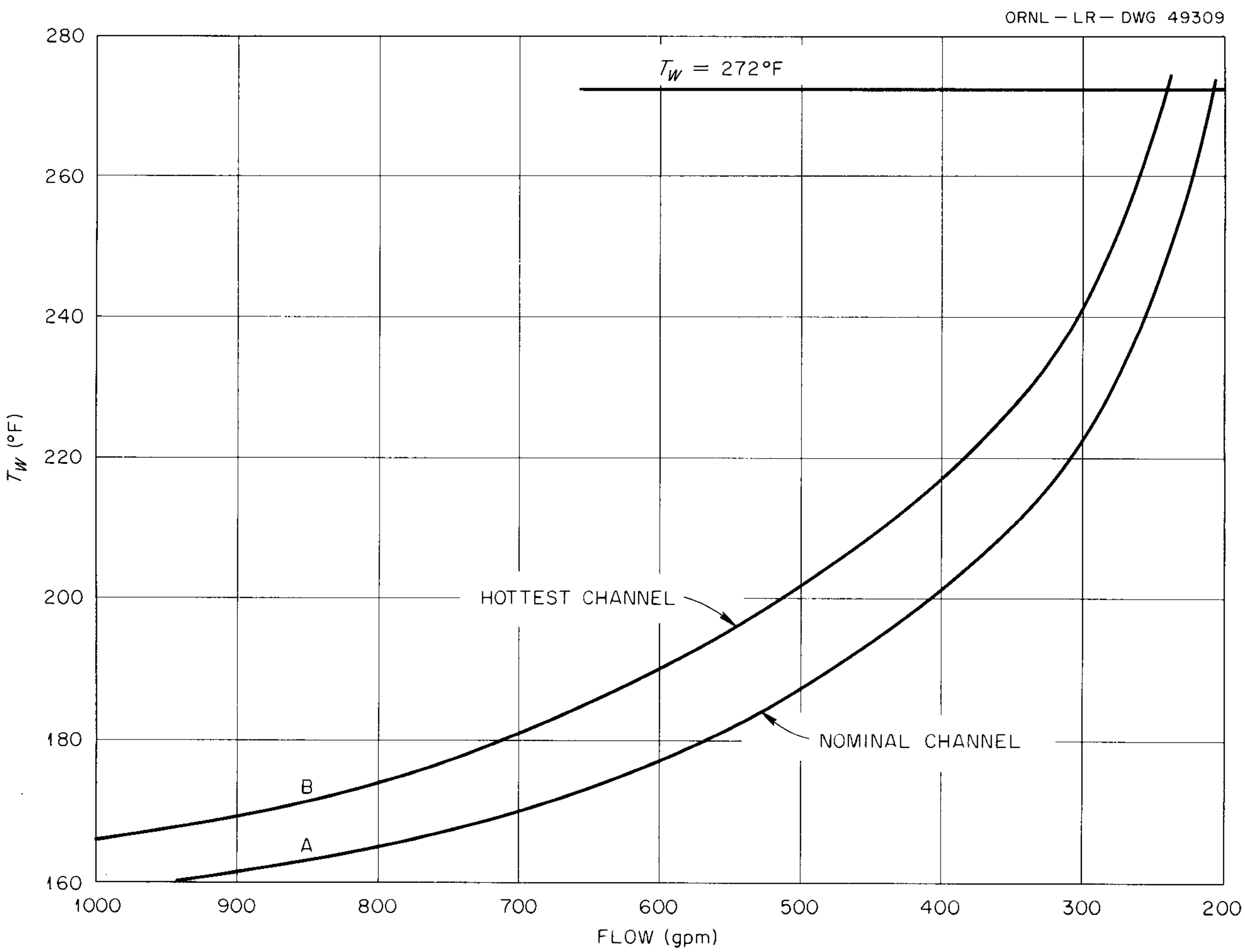

Fig. 7.43. Wall Temperature vs Flow Rate.

8. INSTRUMENTATION AND CONTROL

8.1 Introduction

The ORR safety and control systems have been designed to provide for safe and orderly operation of the reactor from a central control room. Essentially all routine operations, including startup and shutdown, can be monitored and/or controlled from this location.

The control system is designed to relieve the operator of routine manipulations by enabling the instruments which sense changes in the system parameters also to initiate the required corrective action. This approach is consistent with the philosophy of using the operator to supervise the functions of the control system rather than to include him as an integral part of it. Nevertheless, certain actions are required of the operator. In particular, any increase in reactivity beyond that allotted to the power regulation system will require concurrence of both the operator and the control system. The safety system is designed to seize the initiative from both the operator and the control system and to initiate immediate corrective action should any of the significant operating parameters indicate the onset of an unsafe condition. Because of certain features incorporated into the control system, the safety system is very infrequently called upon; however, it is independent of the control system and capable of very fast response when needed.

The ORR control system and instrumentation were designed and safety limits determined after careful analyses of the reactor parameters, such as fuel, shim-rod worth and motion, moderator, and coolant. This design resulted in the use of instrumentation similar to that installed in both the LITR and the MTR, and, in addition, contains a provision for an automatic start mode. Automatic power reduction is provided by:

1. Setback – controlled insertion of the servo-controlled rod.
2. Reverse – simultaneous motor-driven insertion of all shim rods.
3. Slow scram – simultaneous release of all shim rods by using relay contacts to cut off the electrical power supply to the magnet amplifiers.
4. Fast scram – simultaneous release of all shim rods by electronically reducing the output current from the magnet amplifiers as the voltage of the sigma bus increases. (Sigma-bus voltage is increased by any of three independent power-level safety channels as the reactor power increases above 100% of full power or by one log N period channel as the reactor period decreases. When the reactor power reaches 150% of full power or when the reactor period reaches 1 sec over the entire range of operation, the output current from the magnet amplifiers is reduced to the point at which the shim rods are released, shutting down the reactor.)

Raising the power of the reactor from source level to N_L on the log N scale (i.e., 1% of full power) under surveillance of the fission-chamber counting-rate channel can be done either manually or by using the automatic start mode. Power increase from 1 to 100% of full power with level safety and log N period protection can be accomplished either manually or by using the servo-control system.

Process instrumentation monitors reactor coolant flow, reactor differential temperature, and reactor outlet temperature. Signals from these monitoring instruments initiate a power reduction by setback, reverse, or slow scram, as dictated by the degree of the error signal. Reactor-coolant temperature control during a reactor power increase is automatic by step-function control of the reactor cooling system.

The experiments are provided with reactor power-reduction circuits, if required. These circuits are capable of initiating a setback, reverse, or slow scram if an experiment parameter should exceed preset control limits.

Although the basic concepts, criteria, and operation of the above-mentioned controls or protective instrumentation have not changed to any large degree since their original design and installation, continuous reevaluation of reactor controls has prompted revisions and additions to the overall instrumentation of the plant.

8.2 Nuclear Instrumentation

8.2.1 Introduction

Starting at the source level, the fission-chamber counting rate channel (CRM) is employed up to a power level of approximately 10^{-5} N_F. From about 10^{-5} N_F and continuing through 150% N_F, current-measuring channels and gamma-compensated ion chambers are used. Additional current channels working from uncompensated ion chambers are used in the $10^{-2}\ N_F$ range solely as safety devices. In addition to these nuclear instruments the ORR uses other instruments for measurements involving detection of neutrons and gamma radiation.

8.2.2 Safety System

The ORR safety system was designed in accordance with a principle originally established by Newson[1] for the MTR and which has been applied to all ORNL reactors. This principle assumes a startup accident, originating at the source level, brought about by the catastrophic failure of the startup instrumentation, the control system, including interlocks, and manual operation. This failure is postulated to leave the reactor supercritical with the shim-safety rods withdrawing simultaneously at their maximum rate. Except for intrinsic shutdown mechanisms, the reactor safety system remains the only means of stopping the reactivity addition and of turning the excursion by releasing the shim-safety rods. The minimum required performance of the safety system is therefore the protection of the reactor core in the event of catastrophic failure of the startup instrumentation and control system.

The safety system has been used in many ORNL reactors and elsewhere, and its characteristics are well known. Usually, as in the ORR, three channels are so arranged that action of any one of the three can drop the shim-safety rods. The time response from first exceeding the preset limits to the first motion of the shim-safety rods is maintained at approximately 10 msec (release time). This includes magnet-flux decay time with magnet current sufficient to support twice the rod weight. Extensive operating experience has been obtained with the ORR system, as well as with identical systems in other reactors.[2]

The ORR system has been tested in the SPERT facility.[3] As far as can be determined, this is the only safety system which has been required to intervene in a series of excursions, deliberately initiated, wherein a single failure to respond would have resulted in unacceptable damage to the core. The existence of a safety system of this proven high performance reduces to a minimum the consequences of failure of the remaining instrumentation and control systems.

The startup instrumentation and reactivity control system is independent of the level-safety system and has no safety function. The startup instrumentation and control system does, however, include features intended to reduce the number of scrams made necessary by nonsafety failure.

The safety system is also independent of the automatic control system. Although many safety features are included in the action of the control system, it is the function of the safety system to override the control system and quickly shut the reactor down should any of the parameters affecting safety exceed preset values. A block diagram of the safety system is shown in Fig. 8.1.

Although there is redundance in the level-safety fast-scram circuits, coincidence is not required to produce a scram. Each amplifier is driven from an independent neutron-sensitive chamber, and each is capable of shutting the reactor down. The redundance does, however, permit deenergizing and removing for repair any one channel during operation, since it is considered that two operating channels are sufficient to maintain the required reliability for safe operation. It should be made clear that the removal of a level-safety channel is not accomplished by a simple arrangement of manually or automatically operated switches. Such removal and reinsertion action requires close coordination by both Operations and Instrumentation and Controls Division personnel, as well as carefully planned and supervised manipulation of the components and electrical connections. Although the electronics and recorders of each system can be replaced during operation, such is not true of the safety chambers.

Two magnet amplifiers are provided for each shim rod and are electrically interlocked so that either will sustain the required normal magnet current upon the complete or intermittent failure of the other.

There was initially only one log N period channel available for use in the fast-scram circuit; but it was soon recognized that failure of this channel could seriously lengthen a reactor shutdown if the failure were to occur during, or immediately prior

[1] H. W. Newson, *The Control Problem in Piles Capable of Very Short Periods*, ORNL-1857 (Apr. 21, 1947).

[2] E. P. Epler, *Operating Experience with Coincident vs Non-Coincident Reactor Safety Systems*, ORNL-TM-738 (Dec. 12, 1963).

[3] J. R. Tallackson *et al.*, *Performance Tests of the Oak Ridge National Laboratory Fast Safety Systems*, ORNL-3393 (Sept. 12, 1963).

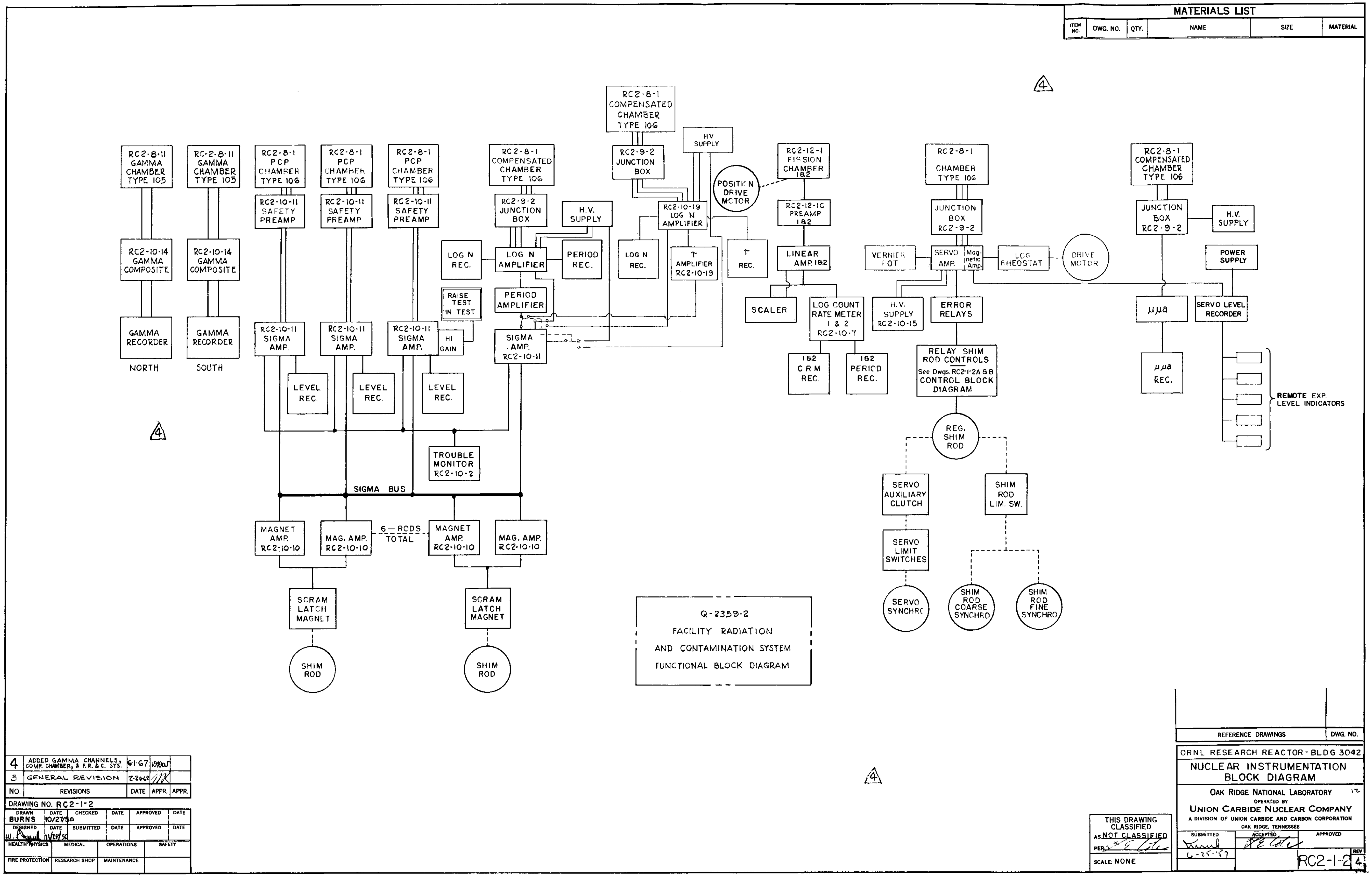

Fig. 8.1. Safety System Block Diagram.

to, a reactor startup. A spare channel, from ionization chamber to the period amplifier including the recorders, was added. This channel is not used to obtain safety redundancy but to minimize reactor downtime due to failure of a single channel. Either of the two channels may be connected into the control circuitry at the operator's discretion. Lights are provided in the control room to indicate the channel in use, and various circuit features monitor the channel in use to ensure that the desired reliability to initiate a scram due to a short reactor period is retained.

In Sect. 8.3, the low-flow operating condition is mentioned. It has been ORNL policy to avoid adjustable safety trip points; but when it became necessary to employ a two-level trip, each level-safety channel was provided with an individual method of switching which is automatic and independent of operator error. This low-level trip safely limits the power to 1.5% of normal full power when the reactor is being operated for special low-power tests. Analysis has shown that power rises on periods as short as 10 msec are handled equally well at both the high- and low-level scram settings.

During the eight years of operating experience, the ORR instrument and control system has not once failed so as to require intervention of the safety system. Also, during eight years of ORR operation the safety system has not once been found to be incapable of protecting the reactor should the instrument and control system have allowed an excursion to develop. It is therefore established that the probability of failure of the instrument and control system is small and that the probability of failure of each of the three safety channels is also small. The probability of simultaneous failure of the independent safety and instrument and control systems becomes the product of four small probabilities, so that the probability of an uncontrolled excursion by this mechanism becomes very small indeed.

8.3 Control System

One of the main objectives of the ORR operation is to continuously maintain the reactor at the highest constant power level which is consistent with safety and the available coolant flow. To accomplish this, heat power has been chosen as the basic control parameter, although it is used indirectly. The instruments which measure flow and temperature difference are quite accurate but are characterized by an inconveniently long response time – several seconds. Even though an exceptionally fast response time is not required of the control system, it is desirable that it be able to initiate corrective action sufficiently rapidly so that fast safety action will be only rarely necessary. The required speed of response is obtained by using the accurate, but delayed, heat-power information to continually calibrate neutron-flux measuring devices by adjusting them to agree with the heat-power information.

8.3.1 Special Features

The control system relays and interlocks are designed to accept information from both the operator and the instrumentation and to impose certain limitations on the use of this information, thus providing an orderly sequence for startup and operation of the reactor. The control-system block diagram is shown in Figs. 1.7 and 1.8. No attempt is made in this section to describe the routine features, such as limit switches, the utility of which should be obvious; however, the significant instrumentation is discussed in detail. A complete relay listing and the elementary diagrams are found in Appendix D. These can be used for a more thorough study of the control system.

The features found in ORNL control systems for reactors of the ORR type include the following:

- Instrument start
- Automatic fission-chamber positioning
- Raise-clutch mode
- Automatic shim-rod-drive rundown
- Servo system
- Key switch

Instrument Start. – While two modes of starting and operating the reactor are provided in the control system (manual and instrument), only the latter is unique to this system. Instrument control is initiated by the operator pressing the "instrument-start-request" push button. The servo starts immediately, and the regulating rod withdraws to its withdraw limit, since the reactor power is below the demand set point. This establishes the "instrument-start" condition. All shim-safety rods begin withdrawing at intermittent speed until a transient 30-sec positive period is detected by the counting channel, at which point withdrawal is stopped. The selection of 30 sec for the minimum operating period is arbitrary. Shorter periods are practical for automatic start but may be troublesome for the operator if the manual start mode is being used. The startup time saved by using shorter periods is too small an amount to be of any significance.

Automatic Fission-Chamber Positioning. – After the rod withdrawal stops, the period grows longer than 30 sec, and rod withdrawal resumes. The process is repeated, however, with shorter and shorter "run" and longer and longer "stop" intervals until a steady 30-sec positive period is established. The power rise continues, and on passing a level corresponding to 8000 counts/sec on the CRM, two changes occur. First, the fission chamber withdraws until it reaches a neutron flux corresponding to something less than 100 counts/sec (this results in the counting channel being readjusted to an operating point near the lower end of its operating range), and second, the control is placed in "mid-range lockout" (MRL), which prevents further automatic rod withdrawal. In addition, rod withdrawal is blocked as soon as 8000 counts/sec is reached, and the blockage continues until the fission chamber stops moving. None of these changes affect rod positions, so the reactor power continues to rise on a 30-sec period until at log $N > 10^{-5} N_F$ the MRL becomes inactive. Except for period permissives, instrument control shifts from the counting to the log N channel. If, after MRL becomes inactive, the period grows longer than 30 sec, all rods will withdraw again as needed to shorten the period. Instrument start is terminated when the power reaches N_L. Instrument start is also terminated instantly if a reverse occurs or if any shim-safety rod drops. Both actions are called to the operator's attention by the annunciator system.

Automatic shim-rod withdrawal is not permitted once instrument start is terminated. During a long run, if the servo rod withdraws to the withdraw limit, an alarm is sounded, and the operator has to correct the situation by withdrawing one or more shim-safety rods a small amount. The servo system, however, is permitted to insert all shim-safety rods if full regulating-rod insertion is insufficient to limit reactor power.

Automatic fission-chamber positioning is a necessary part of the instrument-start system. It is a convenience, however, when starting the reactor manually and is used routinely. In its automatic mode the system repositions the fission chamber as needed to keep the counting channel within its most reliable operating range. Following a reactor shutdown, the chamber is manually inserted until at least 2 counts/sec is detected. When an instrument start is begun, the chamber is inserted automatically until at least 20 counts/sec is detected. When the counting rate reaches 8000 counts/sec (upper end of range), the chamber withdraws until the lower end of its range (100 counts/sec) is reached. An undesirable result of the operator inadvertently inserting the chamber would be the generation of an apparent short period, which could block rod withdrawal or initiate a reverse, if short enough. Inadvertent chamber withdrawal, however, would interfere with an instrument start while the reactor is under control of the counting channel and is therefore prevented. The chamber-drive control may be placed in a manual mode, but doing so blocks the instrument-start mode of the reactor.

Raise-Clutch Mode. – The "raise-clutch" mode permits the shim-safety-rod drives to be "withdrawn" for testing purposes without withdrawing the control rods. When the manual selector is turned to establish this mode, a slow scram is effected. This de-clutches the rods and permits the drives to be withdrawn without moving the rods. There is a check on this because the rods must stay seated or drive withdrawal is immediately stopped.

Automatic Shim-Rod-Drive Rundown. – The "automatic shim-rod-drive rundown" is a convenience for the operator. Whenever the rod and its drive become separated the drive automatically inserts until the magnet contacts the rod. This feature is blocked when the control is in the "raise-clutch" mode to permit drive withdrawal.

Servo System. – The servo system relieves the operator of the tedium of holding the reactor power at the desired level and, in fact, holds reactor power closer to the control point than the operator can. It must be on and operating before the instrument-start mode can be established. It cannot thereafter be turned off without first going out of "instrument start." The servo may fail in such a way that it withdraws the regulating rod fully, placing the reactor on a positive period. If the operator does not discover this and take corrective action, it will be done for him by the log N and level-safety channels through the medium of a reverse. The servo is turned off by the reverse and can be turned back on only by the operator.

Key Switch. – The key switch is an aid to administration and is used to prevent unauthorized movement of rod drives during reactor shutdowns. When the switch is off, the reactor is scrammed, and power to the magnet amplifiers and the withdrawal circuit is disconnected. The switch can be turned on only when the key is in its proper position, and, further, the key cannot be removed until the switch is turned off. Rod-insertion circuits are not turned off and, in fact, are arranged so that insertion requests always take precedence over those for rod withdrawal.

8.3.2 Modes of Operation

Two modes, "start" and "run," are provided. Changeover from the "start" to the "run" condition is made manually when the reactor reaches N_L, that is, ~1% of full power. This represents the point at which the heat-power instrumentation starts to yield reasonably accurate information. Moreover, ~1% of full power is the level above which the cooling requirements exceed that provided by the shutdown coolant system, described in Sect. 6.

In addition to the normal sequence required for routine startup and operation, a submode to the "start" mode of operation is available in order to provide for versatility and to minimize the necessity of making temporary changes to the system which must later be corrected. This mode permits operation up to $\sim N_L$ with no coolant flow.

Low-Flow Conditions. – Under these conditions of operation the reactor can be started up without coolant flow. This permits evaluation of reactor fuel loadings and reactivity effects of experiments installed in the ORR with the reactor vessel open to the surrounding pool. The mode is referred to as the "test" mode. The power is limited by the safety system, and the only control instrumentation affected is the bypassing of the flow and differential-pressure interlocks. The functions of the safety system are discussed in Sect. 8.2.

Normal Conditions. – When the coolant system is functioning normally and full flow is established, startup can proceed either manually or automatically. The startup can continue as long as the startup instrumentation is operating properly, as indicated by a set of confidence contacts, and is indicating a period of 30 sec or longer.

Initially, the shim rods are withdrawn simultaneously only intermittently. This intermittent withdrawal continues until log N confidence (>0.001 on the log N recorder) is reached. At this time the withdrawal can be continuous, because the withdrawal rate is restricted by the reactor period interlocks.

When the upper limit of the "start" mode is reached (N_L), the reactor power level remains at N_L until the operator manually requests a change. After a delay sufficient to allow the operator to assess that the startup can proceed, the "run" condition may be requested and obtained. The operator may then raise the reactor power by increasing the setting of the servo demand. The servo system will then withdraw the servo-controlled shim rod and thus increase the power of the reactor until it equals the demand.

8.3.3 Shim-Rod-Drive Mechanisms

The safety release mechanisms are of the same general type as those used in the High Flux Isotope Reactor (HFIR) and the Engineering Test Reactor (ETR) (i.e., the ball-latch type), as shown in Fig. 8.2. This mechanism consists of a latch head located near the top of the drive tube, in the reactor vessel, with a latch pushrod extending downward into the subpile room through the center of the drive tube; a seal between the pushrod and the drive tube is provided at the lower end of the drive tube. This latch pushrod is moved upward relative to the drive tube in order to engage the latch and is held in position against the force of a release spring by an electromagnet. Shutoff of the magnet current results in release of the latch mechanism. The latch head consists of an outer cage with the same outer diameter as the drive tube. This cage has holes through which a circle of steel balls may be partially extended by the tapered surface of the latch plunger, which is located inside the cage and which is attached to the upper end of the latch pushrod. When the balls are extended, the latch is cocked and is held in position by the electromagnet.

Concentric with the drive tube, but entirely within the reactor vessel, is the shim rod. The top of the drive tube attaches to the bottom of the shim

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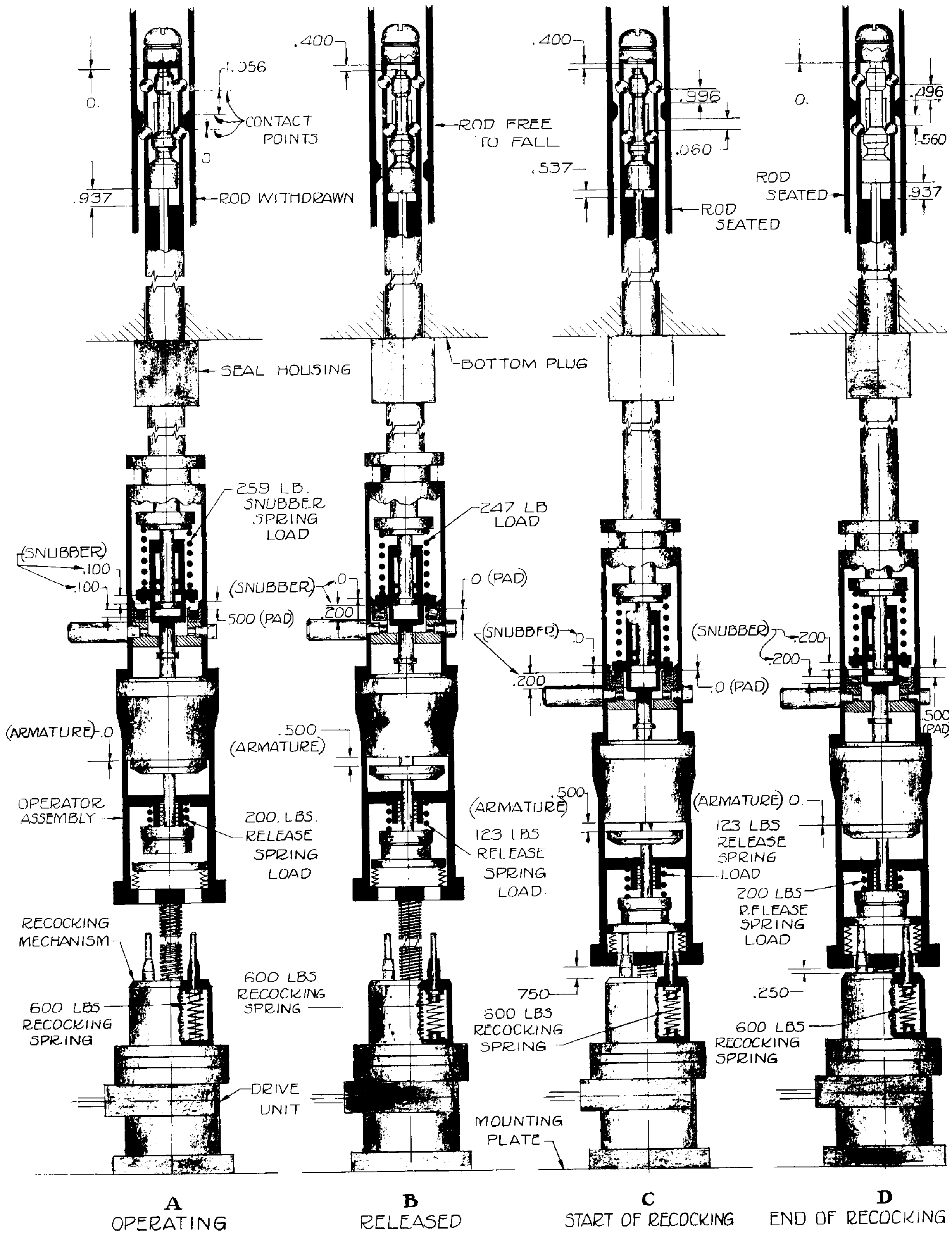

Fig. 8.2. Shim-Rod Drive Mechanism.

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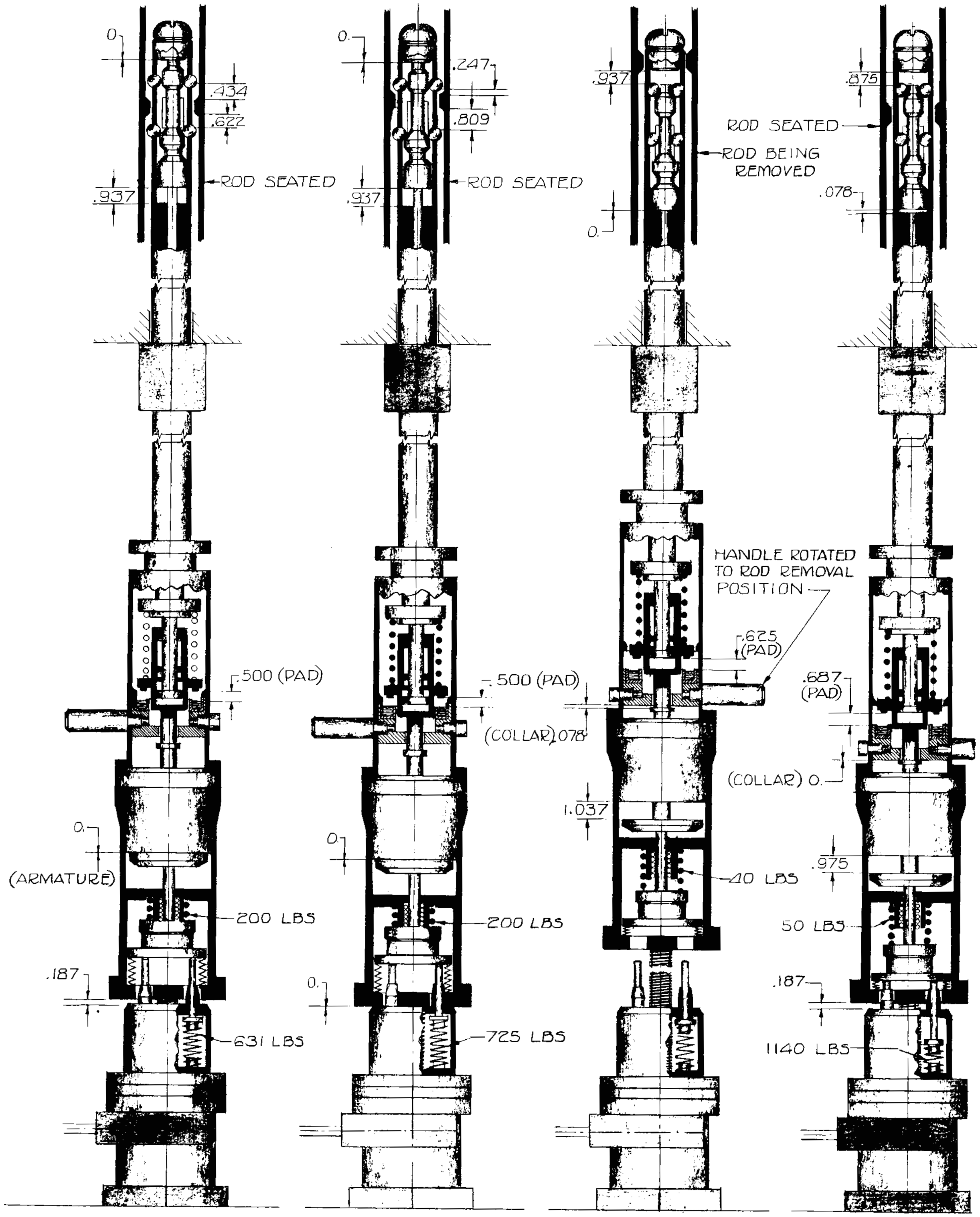

E
LOWER LIMIT SWITCH

F
FULL OVERTRAVEL

G
ROD REMOVAL

H
ROD REMOVAL LIMIT SWITCH

rod by means of a self-aligning coupling referred to as the "floating piston." Inside this floating piston, a shoulder is provided which has an inside diameter slightly larger than the drive tube but smaller than the diameter of the circle of extended balls. Since the latch is cocked while the balls are below the latch shoulder of the floating piston, an upward movement of the drive tube will lift the piston and thus raise (withdraw) the shim rod. If, while the shim rod is raised, the magnet is de-energized, the latch pushrod is forced downward by springs, the balls retract, and the shim rod is free to fall, reducing the reactivity.

The floating piston at the lower end of the shim rod is also a shock-absorber piston which enters a shock-absorber cylinder near the lower end of the stroke to decelerate the shim rod following a scram.

The drive mechanisms are of the type using a nonrotating lead screw of the acme thread type with a rotating nut. The lead screw angle is chosen so that the drive is self-locking; that is, a force applied to the lead screw will not result in the rotation of the nut. The nut for each drive is driven through a suitable gearing by ac motors of the two-phase reversible type. All drives are designed for the same rate of travel, since it is desired to operate them in unison. The rate of travel is 5 in./min with the drive motors operating steadily at their loaded speed.

All drives are provided with position-sensing and transmitting equipment, which is used to transmit information to the control room for the operator. In addition, a variety of limit switches is provided as necessary to indicate drive position and to limit drive motion as required. Seat switches are also provided to indicate when the shim rod is fully inserted. These are independent of the drive mechanisms.

8.4 Process Instrumentation

As explained previously, certain portions of the ORR process instrument system have significant control functions and might well have been included in the preceding sections. However, for convenience and because they are nonnuclear in character, they will be discussed here. These process instruments include devices for monitoring the flow, temperatures, and pressures of the coolant.

In general, the process instrumentation has to do with the heat dissipation from the reactor. For this reason, the control system mentioned in Sect. 6 will be discussed more thoroughly.

8.4.1 Cooling-Tower Control System

Heat absorbed by the cooling system from the reactor is transferred to the atmosphere by the cooling tower (Fig. 8.3) when water from the secondary cooling loop is made to cascade down through the tower against a forced upward draft of air. The flow of air is maintained in the two-cell tower by a two-speed fan in each cell. Each fan can be operated independently, thus providing five steps of control for the two cells

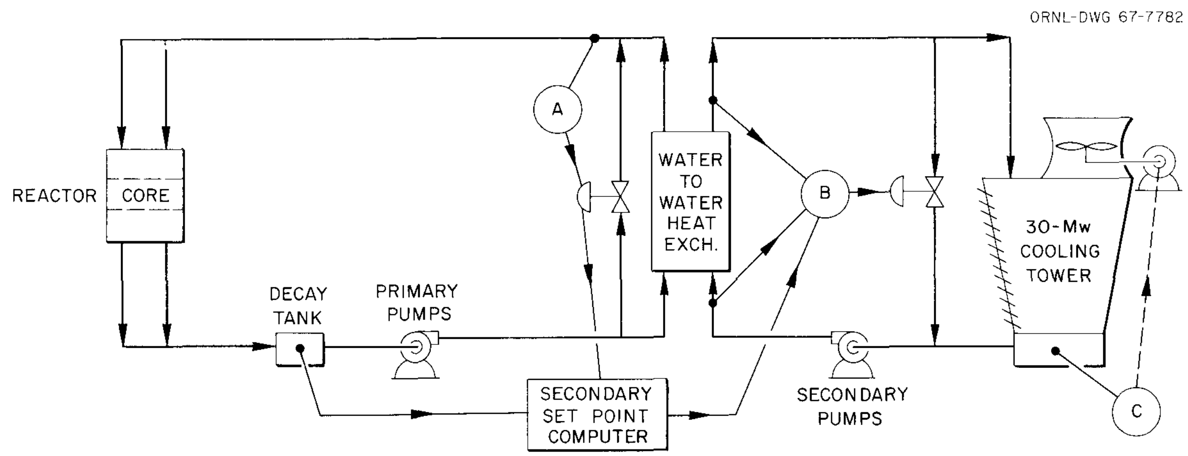

Fig. 8.3. Basic Flow Diagram with Three Temperature Control Loops Shown.

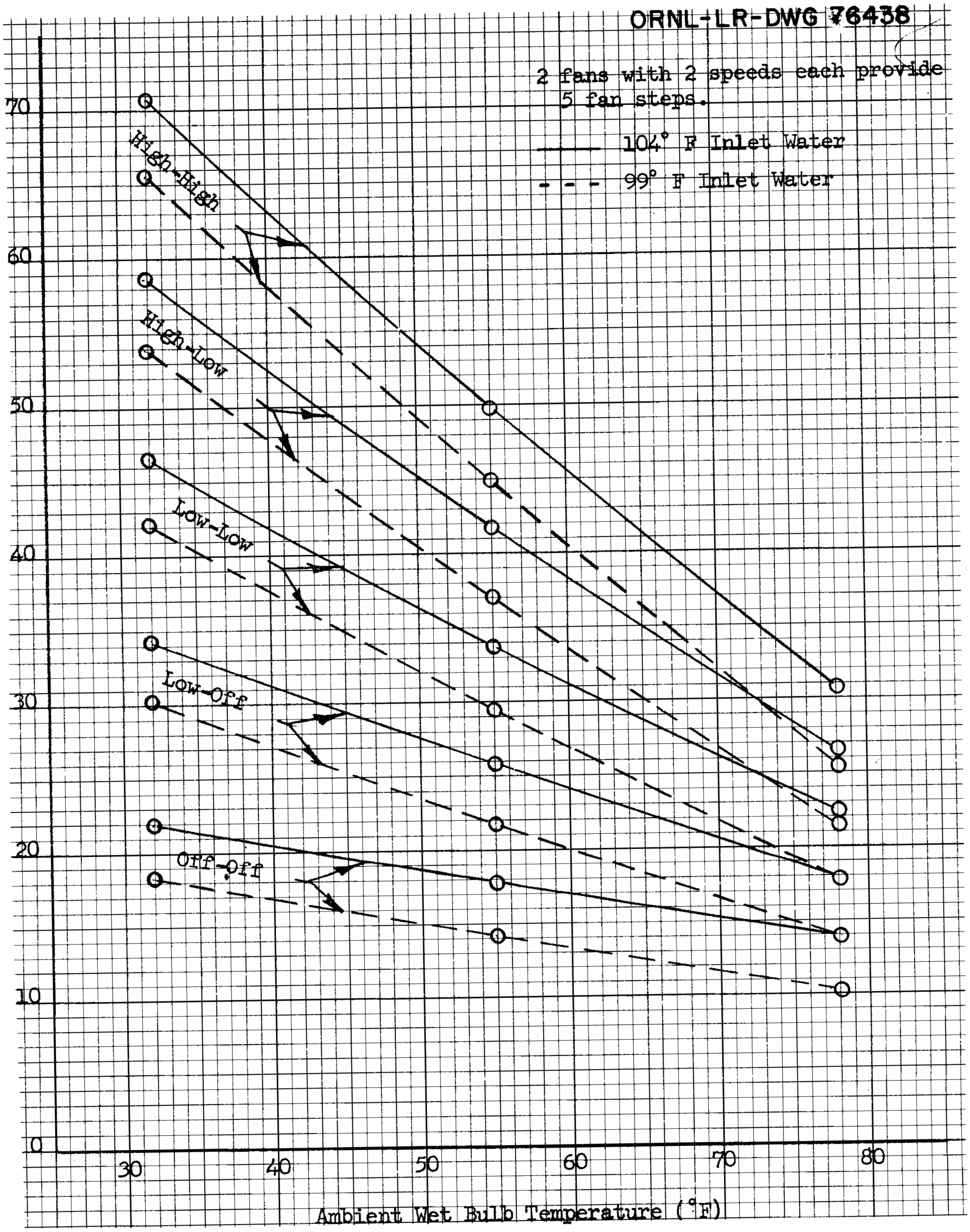

Fig. 8.4. Cooling Tower Capacity at Full-Rate Water Flow.

considered as a unit, that is, off-off, low-off, low-low, high-low, and high-high.

The heat dissipating capacity of the cooling tower is a function of several parameters. An approximate formula as derived from data furnished by the manufacturer is as follows:

$$\mathrm{Mw} = \%F_w(0.35T_i - 0.69T_{wb} + 34.4)(\%S_f) + (0.72T_i - 0.18T_{wb} - 47.06) ,$$

where

Mw = heat dissipation in megawatts (rated at 30.78),

$\%F_w$ = percent of rated water flow over the tower (rated at 10,500 gpm),

T_i = inlet water temperature to the tower in °F (rated at 104°),

T_{wb} = ambient air wet-bulb temperature in °F (rated at 78°), and

$\%S_f$ = percent of full fan speed (both fans on high).

This formula is correct when the parameters are at or near rated values for the tower. How nearly correct it is when the parameters differ widely from rated values is not known. The formula, therefore, should not be used for evaluation of tower performance but may be used in understanding tower operation. Figure 8.4 shows curves of the expected cooling capacity as derived from this formula. From these curves it is shown that, at full-rated water flow, 104°F inlet water temperature, and an ambient wet-bulb temperature of 78°F, the tower cooling capacity may be varied from 13 to 30 Mw by varying the fan speed. With ambient wet-bulb temperature at 55°F, the range is 18 to 50 Mw via fan speed, and with the ambient wet-bulb temperature at 32°F, the range is 22 to 70 Mw via fan speed.

The tower control system maintains the temperature of the water in the cooling-tower basin at a relatively constant temperature somewhat below that needed by the heat exchangers. This method of operation has several advantages:

1. It reduces the dynamic control range of the secondary and primary control loops, making the tower appear to the secondary control loop as an infinite-capacity heat sink.
2. It prevents transient air temperatures from feeding back into the system and upsetting the overall control.
3. It provides a stored cooling capacity which allows rapid increases in reactor power.

A block diagram of the cooling tower control loop is given in Fig. 8.5, and the cooling control system is shown in Fig. 8.6. A gas-filled bulb (TE48) attached to a pneumatic temperature transmitter (TT48) is used to measure the basin water temperature. A conventional pneumatic controller (TC48) with proportional control action is used to compare the measured temperature with the set point value and provide an output pressure signal accordingly. This output pressure in turn operates four pressure switches (TX48A, TX48B, TX48C, and TX48D), which in turn operate the two fans to provide the control as described above.

8.4.2 Secondary Control System

The secondary control system may be considered as consisting of two main parts: a conventional temperature control system and the pneumatic analog set point computer.

Conventional Control System. – Figure 8.7 shows the conventional temperature control system. The mean temperature of the secondary side of the heat exchanger is controlled in such a way that the proper amount of cooling at the right time is supplied to the primary system. Transients are thereby damped out, and the primary is maintained in its optimum control range.

Since it is impractical to measure temperatures within the heat exchanger, gas-filled bulbs and pneumatic temperature transmitters (TE47A, TE47B, TT47A, and TT47B) on the inlet and outlet water lines are used to provide and transmit the mean readings. A simple computing relay (TM47D) has an output which is the average of the two readings. This is considered as the mean secondary temperature which is to be controlled. A conventional pneumatic controller (TC47) then compares this mean temperature with a set point provided by the set point computer and provides an output control signal to operate the secondary control valve. The valve bypasses outlet water back to the pumps instead of over the cooling tower. By thus controlling the amount of water bypassed, the mean temperature of the secondary system is controlled.

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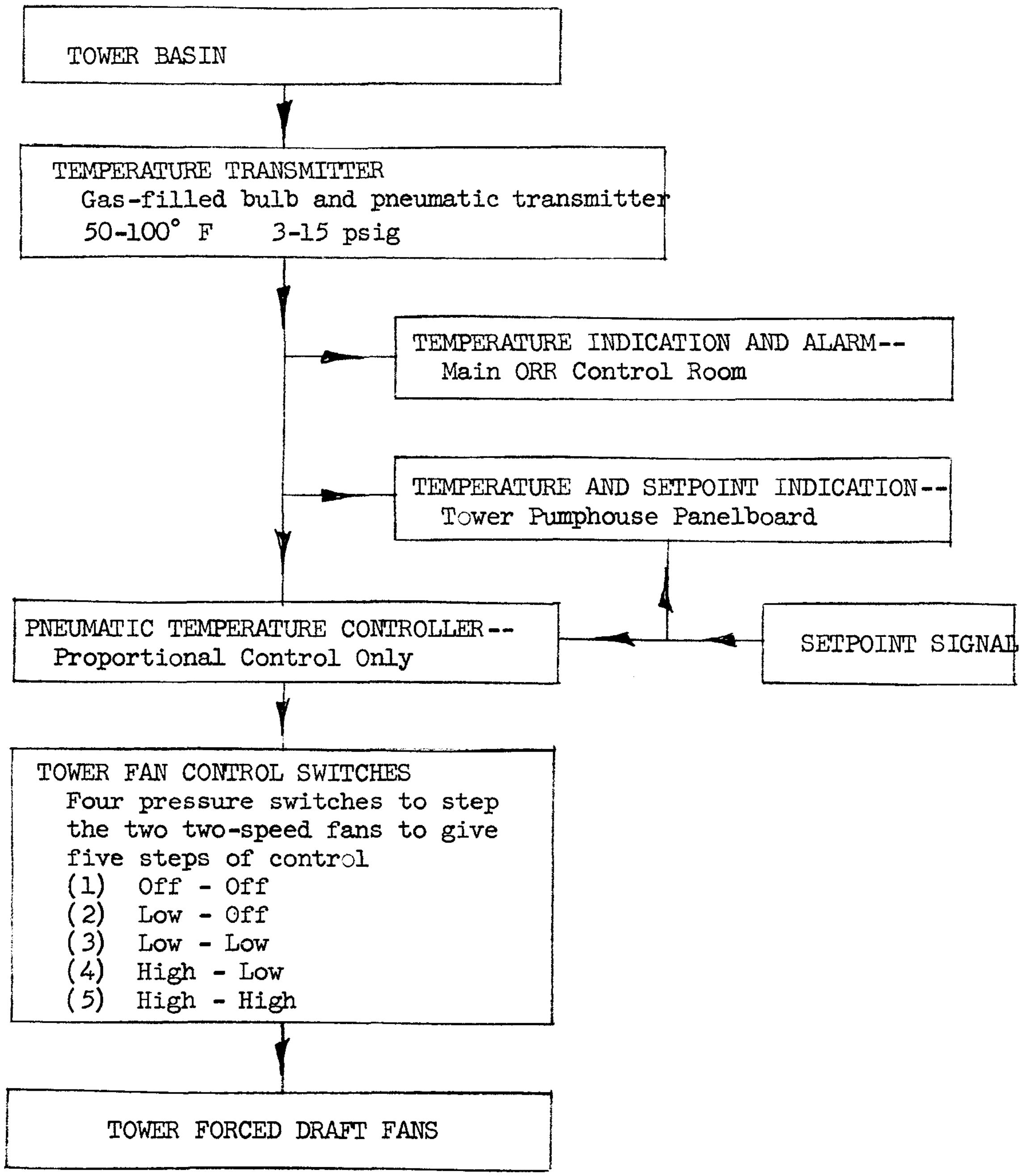

Fig. 8.5. Cooling Tower Control System Block Diagram.

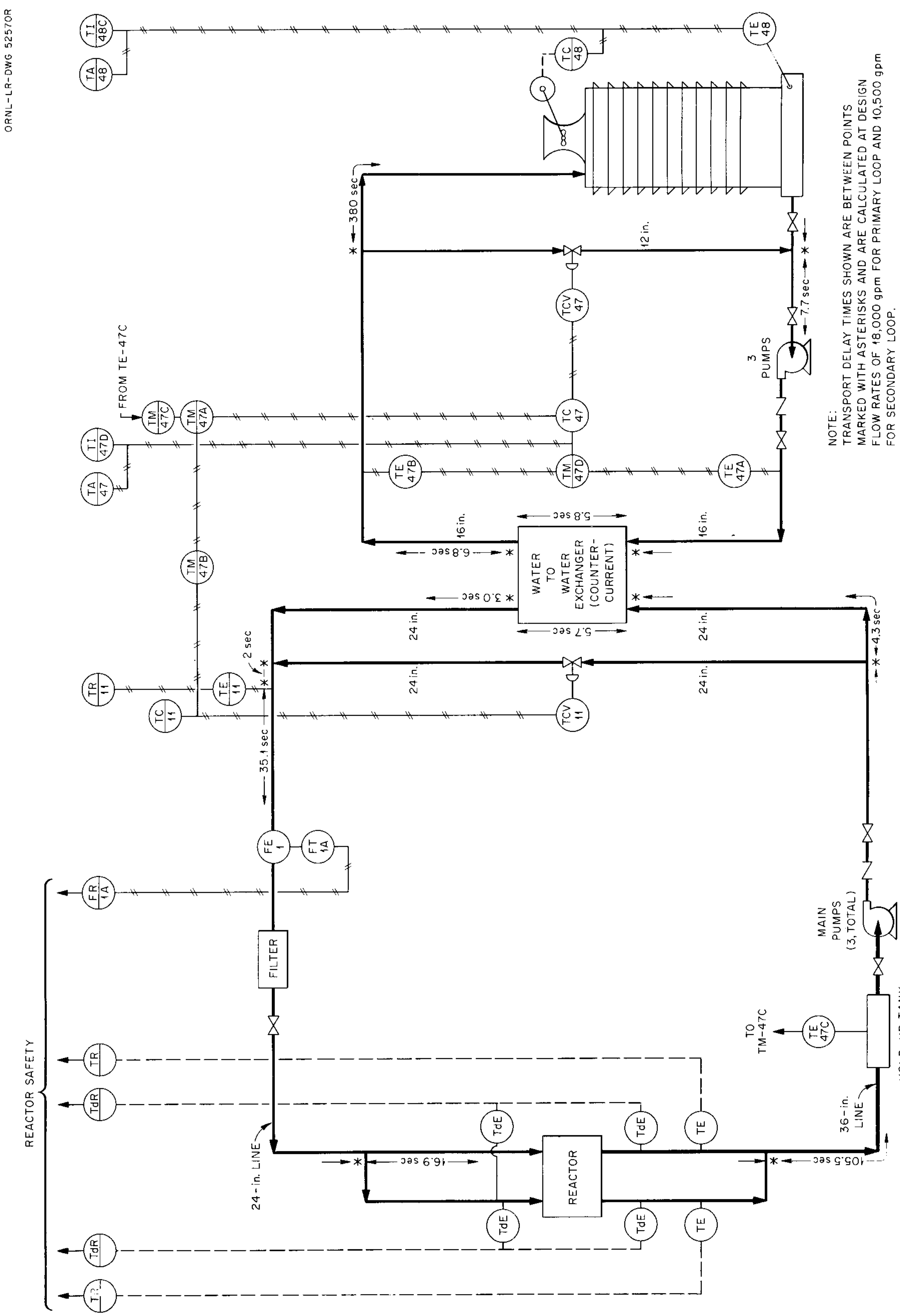

Fig. 8.6. Cooling Control System.

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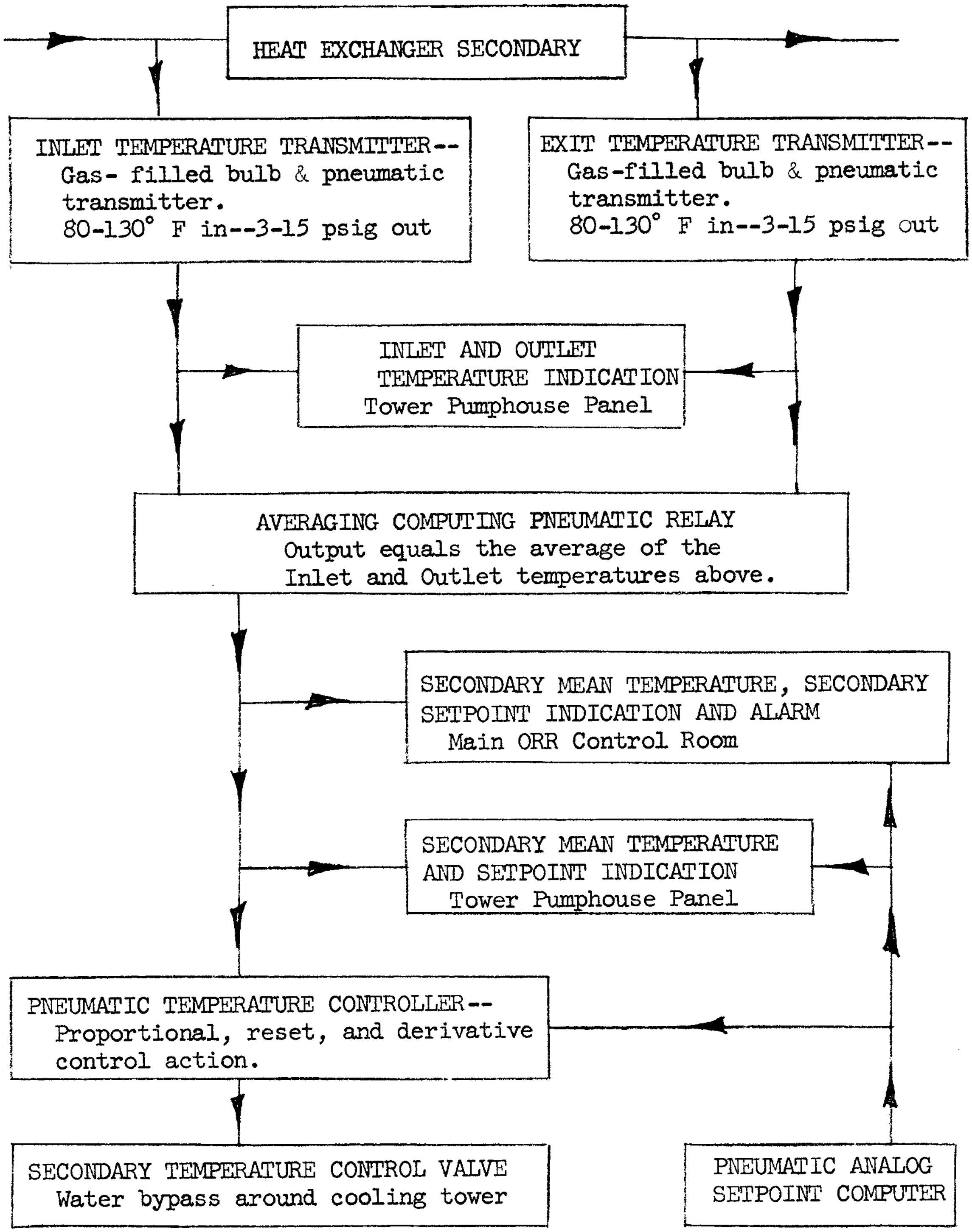

Fig. 8.7. Secondary Coolant Control System Block Diagram.

Set Point Computer. – Figure 8.8 shows a block diagram of the functions of the analog computer used for obtaining the set point for the secondary control system. The set point is determined by two conditions: fluctuations in the reactor power and adjustments in the primary system made to correct these fluctuations. The computer system may be conveniently described by tracing the signals from each of these sources.

Reactor Power Fluctuations. – The temperature of the cooling water leaving the reactor is taken as an indication of reactor power level. The outlet temperature is measured by a gas-filled bulb (TE47C), and a signal corresponding to a 100 to 150°F temperature range is transmitted by a pneumatic transmitter (TT47C). A pneumatic computing relay (TM47C) is designed to take this 100 to 150°F temperature signal C, compare it with a desired temperature, and produce an output signal P corresponding to the corrective action required by the secondary system. In accomplishing this, C is compared with two set points, A and B, and the computer output may be represented as:

$$P = B + \frac{1}{PB}(A\text{-}C) \quad .$$

Under normal operating conditions, both A and B are set for 117 to 120°F. The temperature of the inlet cooling water varies over this range, and at a reactor power level of zero the outlet corresponds to the inlet. Consequently, when the outlet cooling water is, for example, 120°F, or equal to the inlet temperature, the signal C is equal to A and the term $(1/PB)\,(A\text{-}C)$ is zero. The output of the computer, P, is then equal to B, which is the zero-power temperature, and no change is called for from this source in the secondary cooling loop. If the reactor power rises, C increases, and the term $(1/PB)\,(A\text{-}C)$ assumes a negative value. The output P then drops below the set point temperature B and provides a signal to the secondary cooling system calling for more cooling capacity. If the reactor power drops, the reverse is true; P increases, and the secondary is asked to provide less cooling.

Adjustments in the Primary Coolant System. – When a change in water temperature occurs in the primary cooling system, the primary bypass valve responds immediately to provide a corresponding change in cooling capacity. Temperature fluctuations are sensed by temperature elements (TE11A or TE11B) and proportional signals are transmitted by TT11A and TT11B to the valve controller (TCV11). These signals, which determine the valve position, are also picked up by an isolator-booster (TM11B) and fed into the computer system.

The optimum operating condition for the primary bypass valve has been determined as 40% open; this condition is produced by an 8-psi signal from TT11A or TT11B. Consequently, when an 8-psi signal is received, the primary system is at equilibrium, and no corrective action is necessary in the secondary. Equating this 8-psi signal to equilibrium conditions is accomplished by an 8-psi-equivalent spring in the pneumatic computing relay TM47B. An 8-psi signal to the valve is received at A by TM47B and is exactly balanced by the equivalent spring. As the primary valve opens and closes to meet fluctuations around the temperature set point, the output of TM47B will correspondingly swing positive and negative. The relay is adjusted to give a 9-psi output for equilibrium conditions. It may thus make a 6-psi swing in both the positive and negative directions in accordance with the 3-to-15-psi range of the instrument.

Still another pneumatic computing relay, TM47A, sums the two conditions determining the secondary-cooling-loop set point. Since the signal from the primary bypass valve swings around 9 psi, a 9-psi-equivalent spring is provided in TM47A to balance this offset and give a zero signal to the secondary from the primary valve when it is in its optimum condition.

8.4.3 Primary-Coolant Control System

As mentioned previously, the temperature of the cooling water in the primary control system is determined by a conventional temperature control system. Temperature element TE11A senses the reactor inlet temperature at a point on the downstream side of the converging flows from the heat exchangers and the bypass line. The signal from TE11A is converted into a 3- to 15-psi output by TT11A, which is fed through an isolator-booster relay (TM11A) to a recorder (TR11) in the control room and to a controller (TC11) whose output actuates the primary bypass valve TCV11.

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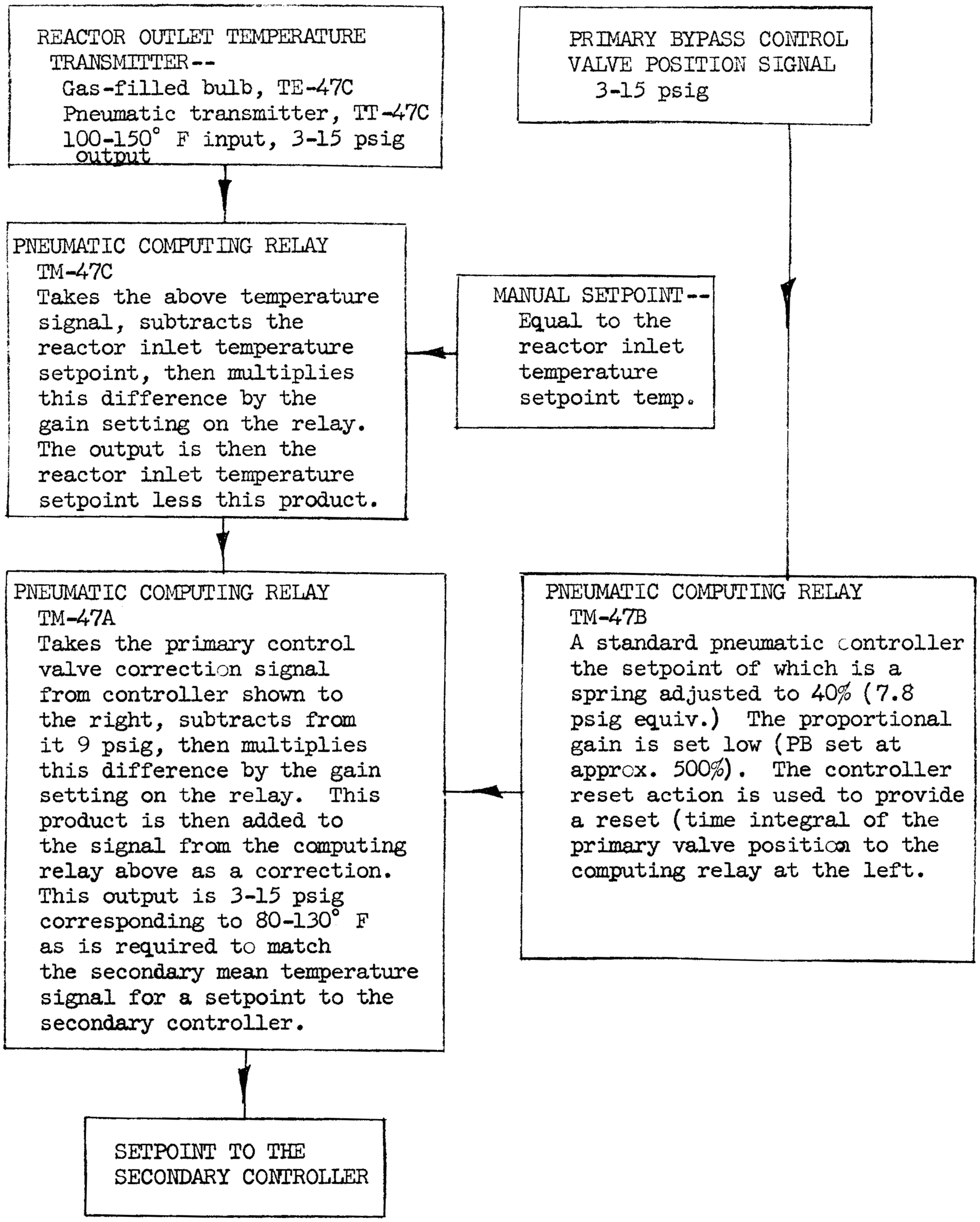

Fig. 8.8. Secondary Coolant Control System Set Point Block Diagram.

8.4.4 Reactor Differential and Outlet Temperature System

Since the increase in the cooling water temperature is proportional to the reactor power level and reflects temperatures within the core, measurements of this parameter are used to indicate the limits of safe operation for the reactor. Two types of temperature measurement, differential and outlet, are connected in the reactor control system, giving alarm, setback, reverse, or scram if either exceeds safe limits.

Cooling water enters the top of the reactor tank through two diametrically opposed lines and flows down through the core. It then leaves the reactor through two other diametrically opposed lines at the bottom. Both inlet and outlet lines are located in close proximity directly beneath the reactor core in the reactor's pipe chase, and all differential and outlet temperature elements are conveniently located there.

Differential Temperature System. – The difference between the temperature of the cooling water entering the reactor and that leaving is a direct indication of the heat or power being produced in the reactor core. The general working formula for reactor thermal power at the ORR is:

$$\mathrm{Mw} = 1.448 \times \mathrm{gpm} \times \Delta T \times 10^{-4} \ ,$$

where Mw is reactor power in megawatts, gpm is cooling water flow in gallons per minute, and ΔT is the temperature differential across the reactor in °F.

At normal 30-Mw operation, the temperature of the water entering the reactor is 120°F, and the corresponding temperature of the water leaving is 131.6°F. This gives a temperature differential of 11.6°F. If this differential rises above prescribed limits, it is an indication that the 30-Mw reactor power level is being exceeded. A ΔT reading greater than 13.0° calls for an alarm, 13.5° calls for a setback and reverse, and 15.5° for a scram.

Since the cooling water passes through the core from two lines, it is likely that changes in core loading will cause differences in the coolant temperatures in the two lines. For this reason, two sets of readings are used to determine ΔT, one from the north and one from the south side. Their average is used as an indication of the overall ΔT parameter and to determine the proper control action. Control action is taken if either channel gives an abnormal indication. The two channels are labeled Td1A and Td1B. The arrangement of components in the Td1A series is outlined in the following paragraph. The arrangement is exactly duplicated in the Td1B channel.

Since the water entering the reactor is at the same temperature in both lines, the inlet temperature elements (TdE1A1 and TdE1A2) for both ΔT measurements are located, for convenience of access, in the south line. On the outlet side, TdE1A3 is in the south line and TdE1A4 is in the north line. All are thermohm or resistance-type temperature elements which change their resistance value in direct proportion to their temperature. In a bridge-type connection, the sum of the voltage drops produced across the two inlet line resistors is subtracted from the sum of the voltages produced across the two outlet line resistors. This difference is proportional to the average ΔT for the two cooling lines and is recorded by a recorder (TdR1A) in the control room. Microswitches on the recorder give the alarm, setback, reverse, and scram actions at the proper levels.

Outlet-Temperature System. – If fuel-plate temperatures within the reactor core exceed 270°F, there is danger of boiling occurring. With allowances for a proper safety factor, it has been determined that the cooling-water outlet temperature should be maintained below 135°F in order to maintain safe core temperatures. Outlet temperature is measured by two thermohm elements (TE43 and TE44) and recorded in the control room by recorders TR43 and TR44. The operating temperature at 30 Mw is in the range 129 to 131.6°F. Microswitches in the recorders initiate an alarm at 134°, a setback and reverse at 135°, and a scram at 140°.

Thermocouples TE2A, TE2D, and TE2G supply readings of inlet and outlet water temperatures to a multipoint recorder in the control room for readout purposes only.

8.5 Radiation Instrumentation

The ORR is one of 17 facilities at the Oak Ridge National Laboratory equipped with networks of radiation and air monitoring instruments connected to the Laboratory Emergency Control Center. The network in the ORR building is also monitored from the ORR control room. Radiation and contamination detection systems are installed in the reactor building to continuously and automatically determine the radiation condition of the entire facility and to relay this information to the central control panelboard in the ORR control room. Should the radiation level or the air activity in a large portion of the building exceed preset values, an audible alarm in the building is actuated, warning lights outside the building flash, and an alarm signal is transmitted to the Emergency Control Center, Building 2500.

8.5.1 Description

The beta-gamma radiation level is monitored by 15 monitrons located throughout the building and and in the reactor cooling-water pump house (Table 8.1). Air is monitored for beta-gamma emitting particles by nine continuous air monitors located throughout the building.

Since none of the health physics functions originally designed into these monitrons and monitors were altered, each instrument is an independent unit that retains all of its local alarm features. Each instrument, however, is connected to an individual indicator module in the reactor control room. By means of three colored lamps which normally give a dim light, an indicator module indicates the condition of the instrument to which it is connected; that is, a white lamp burns at full intensity if the instrument becomes inoperative, an amber lamp burns at full intensity if the "caution alarm level" (7.5 mr/hr for a monitron and 1000 counts/min for an air monitor) is reached, and a red lamp burns at full intensity if the "high alarm level" (23 mr/hr for a monitron and 4000 counts/min for an air monitor) is reached (Table 8.2). A change in intensity of any lamp, that is, from dim to bright, is announced by a buzzer.

The high-level alarm output on each of the six selected monitron indicator modules (Table 8.1) is connected to an evacuation module; six selected air-monitor indicator modules are connected to an evacuation module in a similar manner. If an evacuation module receives a high-level alarm signal from any two or more monitrons or two or more air monitors, the building evacuation system is activated.

The radiation level measured by the six monitrons 1, 2, 4, 6, 7, and 9 and the air activity measured by two air monitors 1 and 8 are continuously recorded on a strip chart in the control room.

Table 8.1. Radiation and Contamination Monitors Installed in the Reactor Building

Location	Instrument Number
Facility Radiation Monitors[a]	
Southeast basement	2
North central, first level	3
Loop 2 area	5
Third level west	11
Southwest balcony	10
Control room	14
Radiation Monitors with Control Room Alarm Only[a]	
Northwest basement	1
Pool side, south[b]	4
First level, northeast	7
Second level north	8
Pump house	15
Second level west	9
Center balcony	13
First level east	6
Top of pool[c]	12
Facility Contamination Monitors[d]	
Basement east	1
Second level northwest	4
Basement northwest	5
First level south	6
First level northwest	7
Pool side	8
Contamination Monitors with Control Room Alarm Only[d]	
Pneumatic tube area	2
Basement south	3
Hot cell	9

[a]All monitors are ORNL model Q-1154B-13 (Sect. 8.5.2b).

[b]Equipped with special ranges of 0.5 and 1 r/hr.

[c]Equipped with special ranges of 25 and 125 mr/hr.

[d]All monitors are ORNL model Q-2240B-4 (Sect. 8.5.2c).

Table 8.2. Central Control-Panel Alarm Indications

Instrument Indication	Lamp Intensity		
	Red	Yellow	White
Normal	Dim	Dim	Dim
Caution level[a]	Dim	Bright	Dim
High level[b]	Bright	Bright	Dim
Instrument trouble or out of service	Dim	Dim	Bright
Instrument removed[c]	Bright	Bright	Bright

[a]Caution level for air monitor is 1000 counts/min and for monitron is 7.5 mr/hr.

[b]High level for air monitor is 4000 counts/min and for monitron is 23 mr/hr.

[c]Lamp intensities exist until a maintenance connection is made giving "out-of-service" indication.

8.5.2 Components

Panelboard. – The central panelboard for the entire system, located in the reactor control room, consists of 12 module racks which contain one indicating module for each radiation detection instrument, a coincidence module for the radiation alarm system, a coincidence module for the contamination alarm system, a buzzer module, and a manually activated evacuation module. The racks and modules are made of anodized aluminum, and the modules have anodized Metalphoto front panels. A Metalphoto text strip is provided at the top of each rack for instrument identification.

Monitrons. – The remote monitrons (ORNL model Q-1154B) are ac-powered null-type radiation detection instruments for monitoring gamma radiation. A monitron consists of two basic units: (1) a control chassis, which contains the power supply, the main amplifier, a radiation-level indicating meter, and the controls; and (2) a preamplifier and ion-chamber detector assembly. The detector assembly can be located remotely from the control chassis. A 0-to-10-mv recorder or a 0-to-1-ma meter can be connected to the monitron. The range of the meter is 0 to 25 mr/hr, and for gamma radiation the calibrated accuracy of the meter is within 3% of the experimentally determined value. The zero setting is adjusted manually at the instrument.

In addition to these standard features, the monitrons installed in the reactor building have two special features: (1) additional relays were installed on the main chassis, and (2) a meter relay was installed on an accessory chassis.

The instrument and alarms operate as follows:

1. A power failure or a disconnected power cord will cause the white lamp on the central control panel to burn brightly. After power is restored and following a 1-min delay, the white lamp can be reset to its normal "dim" mode. During this 1-min delay, caution and high-level alarms at the instrument might sound, but the same alarms on the central panel are locked out and will not sound.
2. At the caution radiation level of 7.5 mr/hr, an electronic alarm circuit will cause the yellow lamp on the central panel to burn brightly and the bell at the instrument to ring. The instrument automatically resets itself when the radiation level decreases below 7.5 mr/hr.
3. The 0-to-1-ma meter output terminals are connected to the accessory meter relay, which is set to operate at 23 mr/hr, the high-level alarm limit. At 23 mr/hr, the bell at the instrument will already be ringing, since it started at 7.5 mr/hr, and the red lamp on the central panel will begin to burn brightly. When the radiation level decreases to less than 23 mr/hr, the instrument is automatically reset by an interrupter (allows a reset every 30 sec) at the central panel. While the instrument is still recording a value in its "caution" range, the same functions described above will be performed.
4. On a dual-range instrument, the white light on the central panel burns brightly to indicate that the instrument is set on its highest range or is inoperative.

Air Monitors. – Two types of air monitors were installed: the mobile air monitor (ORNL model Q-1740) and the beta-gamma continuous air monitor (ORNL model Q-2240).

Mobile Air Monitor. – This monitor is equipped with a logarithmic counting-rate meter, which normally does not require range changing. Any background radiation level within the range of the instrument can be suppressed, and from one to five decades may be selected for display on a strip-chart recorder. This scale arrangement allows the instrument to accumulate air-contamination data and still function meaningfully in areas where the background count is high.

A model H Leeds and Northrup Speedomax recorder with upscale, downscale, and set-point alarms indicates high radiation, instrument failure, and above-tolerance radiation respectively. The local alarm circuit is so arranged that, on instrument failure or very high radiation conditions, an alarm bell and a red annunciator light are actuated simultaneously. If the air contamination increases to a point above the set point, an amber annunciator light, indicating tolerance level, is actuated. The local annunciator lights are mounted on top of the instrument cabinet so that operation of the monitor can be observed from any position in the immediate area.

Beta-Gamma Continuous Air Monitor. – This monitor consists of an aspirating system, a paper-tape filter, a halogen-type Geiger-Mueller tube detector, a linear counting-rate meter, a recorder, and visible and audible alarms. The air flow through the filter is controlled at 3 cfm by a blower. A sample of beta-gamma emitting particles may be collected on the paper-tape filter (collected-sample size of $1 \times 2\frac{1}{2}$ in.). The collection time per sample may be one day or one week, depending on the timer installed, or the tape may be advanced manually at any time. The detector analyzes the radiation level of the sample as it is being collected. The counting-rate meter is a linear duty-cycle type using a single range and having its high-voltage supply as an integral part of the unit. The normal range is 0 to 10,000 counts/min, but a 0-to-25,000-counts/min range may be obtained by relocating two jumper wires under the chassis. The input voltage sensitivity of the rate meter is 200 mv. The overall accuracy of the rate meter, including the effect of long-term drift, is within 5% of correct values. The corona-regulated high-voltage supply is nominally 900 v with $\pm$150-v adjustment, and the maximum load current is 20 μa.

The counting-rate meter has adjustable high-level and caution alarms and puts out a full-scale signal of 1 ma to be used as an input to an integrally mounted Rustrak recorder. The caution alarm is adjustable over the range of $\sim$2 to 58%, and the high-level alarm is adjustable from the caution-alarm set point to the full-scale value. The caution alarm is an electronic circuit which employs a dual triode and plate relay with potentiometer adjustment. The relay is energized below the trip point and is deenergized by current transfer from one triode to the other by a diode that couples the cathode circuits. Hysteresis is $\sim$4% of full scale. The high-level trip is accomplished by the high contact on a contact-making meter. The associated high-level relay is deenergized below the set point. Upon reaching the set point, the meter contacts and relay are locked in and can be released only by the depression of the release push button. A low-level pointer on the panel meter has no contacts and is used only as a visual indicator. The pointer is set at the level corresponding to the caution-alarm set point.

The instrument has an alarm panel with four lights, a bell, and a buzzer. When the caution set point is reached, an amber light comes on and the buzzer is energized. There is no switch to silence the buzzer, and normally the tape is advanced manually when this point is reached. When the high-level trip point is reached, a red light comes on and the bell rings. The bell can be silenced by a toggle switch, and when this is done, an amber neon indicator comes on.

Filter-tape breakage is also indicated. If a tape breaks, a red neon indicator is energized, and the caution circuit is energized through a flasher. The amber neon bulb burns continuously, and the caution light and buzzer come on intermittently. If a tape breaks and the caution alarm sounds at the same time, the tape-break neon light, the caution light, and the buzzer would be on simultaneously and continuously. A test button permits checking the alarm panel by simultaneously simulating tape break and high-level alarm signals.

9. ORR SHIELDING

9.1 General Criteria

As stated in previous sections, the ORR was originally designed for a steady operating power of 5 Mw. However, the shielding was designed to satisfy the permissible radiation dose limitations to individuals set forth in National Bureau of Standards Handbooks No. 59 and No. 63 for a steady operating power of 30 Mw. In unlimited access areas, the shield design is such as to limit the maximum dose rate to 0.75 millirem/hr. This intensity includes the combined effects of all radiations, and, based on a 40-hr week and a 50-week year, it represents 30% of the annual permissible dose of 5 rems recommended by the NBS Handbooks as an acceptable amount for workers handling radioactive materials. In limited access areas, sustained exposure is unlikely and can be controlled, so higher dose rates are permitted there. This fact makes possible some economy in shield design.

Adequate shielding is provided not only for the reactor itself but for portions of the primary coolant loop and for the primary and pool coolant cleanup systems as well. Shielding is also provided where necessary for the various components of the off-gas and all ventilation systems. The demineralizers, filters, degasifier, and other equipment are also located in individually shielded cells or cubicles. Shielding provided for the cells is supplemented in some cases with direct lead shielding on the equipment. Should significant local areas of high radiation be detected, they are immediately reduced by the use of additional local shielding.

9.2 Main Reactor Shielding

The main biological shield for the reactor is the water-filled reactor pool and the concrete walls of the reactor pool. It may be considered to consist of three parts: the top shield, which is of water; the concrete-and-water lateral, or radial, shield; and concrete, steel, and water bottom shield.

9.2.1 Reactor Top Shield

The reactor core is shielded above by water, the surface of which is 23 ft above the top level of the fuel. By direct use of experimental data from the Bulk Shielding Reactor (BSR),[1] shown in Fig. 9.1, the gamma dose rate at the pool surface during operation at 30 Mw is found to be approximately 8 millirems/hr. This dose rate at the water level above the reactor is above that recommended for long-term exposure at a permanent installation. These criteria, however, are not strictly applicable, since the likelihood that someone will remain in that area is quite remote. The neutron contribution to the dose rate is negligible. The ^{16}N generated in the pool surrounding the reactor tank is dispersed as described in Sect. 6.

[1]F. C. Maienschein *et al.*, *Attenuation by Water of Radiation from a Swimming-Pool-Type Reactor*, ORNL-1891 (Sept. 7, 1955).

9.2.2 The Lateral Shield

An analysis[2] of the adequacy of the lateral shield was made at the obviously weaker points of the shield: near the large experiment facilities, beam holes, and the primary coolant lines. The shielding at the large facilities is the weakest part of the shield. When the facilities are in use (drained), supplemental shielding must be provided to reduce the background to below tolerance. With the exception of these facilities, the shielding on the north and south faces of the lateral shield consists of at least 4 ft of barytes concrete, which has proved more than adequate for 30-Mw operation (see Fig. 9.2). Figure 9.3 shows the penetrations in the lateral shield. The thimbles shown in the figure are above the core, so that during operation no radiation problems are encountered on the balcony. However, during refueling, access to the balcony is prohibited, since hot fuel is being transported through the pools; as the element passes the thimbles, the radiation level increases just opposite the thimble.

Figure 9.2 also shows the barytes shielding on the east lateral shield, through which the six horizontal beam holes penetrate. Here the concrete is approximately 8 ft thick. To supplement this material, movable concrete shields are used to provide protection when beam-hole experiments are in progress.

9.2.3 The Bottom Shield

A higher radiation field can be tolerated in this area because there is no need for personnel to remain for long periods of time with the reactor operating at 30 Mw, and a locked door is provided to prohibit unnecessary entry. There is approximately 7 ft of barytes concrete between the subpile room and the reactor tank. This "ceiling" of the subpile room is penetrated by a bottom plug (see Fig. 9.4) which contains the penetrations for the shim-rod drives. It is filled with a mixture of iron shot and barytes concrete. Figure 9.4 also shows the pipe chase walls. Figures 9.5 through 9.9 show the pool walls at different elevations and views so that a more complete overall image of the pool structure can be visualized.

[2]W. Zobel *et al.*, *Review of the ORR Shield for 30-Mw Operation*, ORNL-CF-58-6-13 (June 16, 1958).

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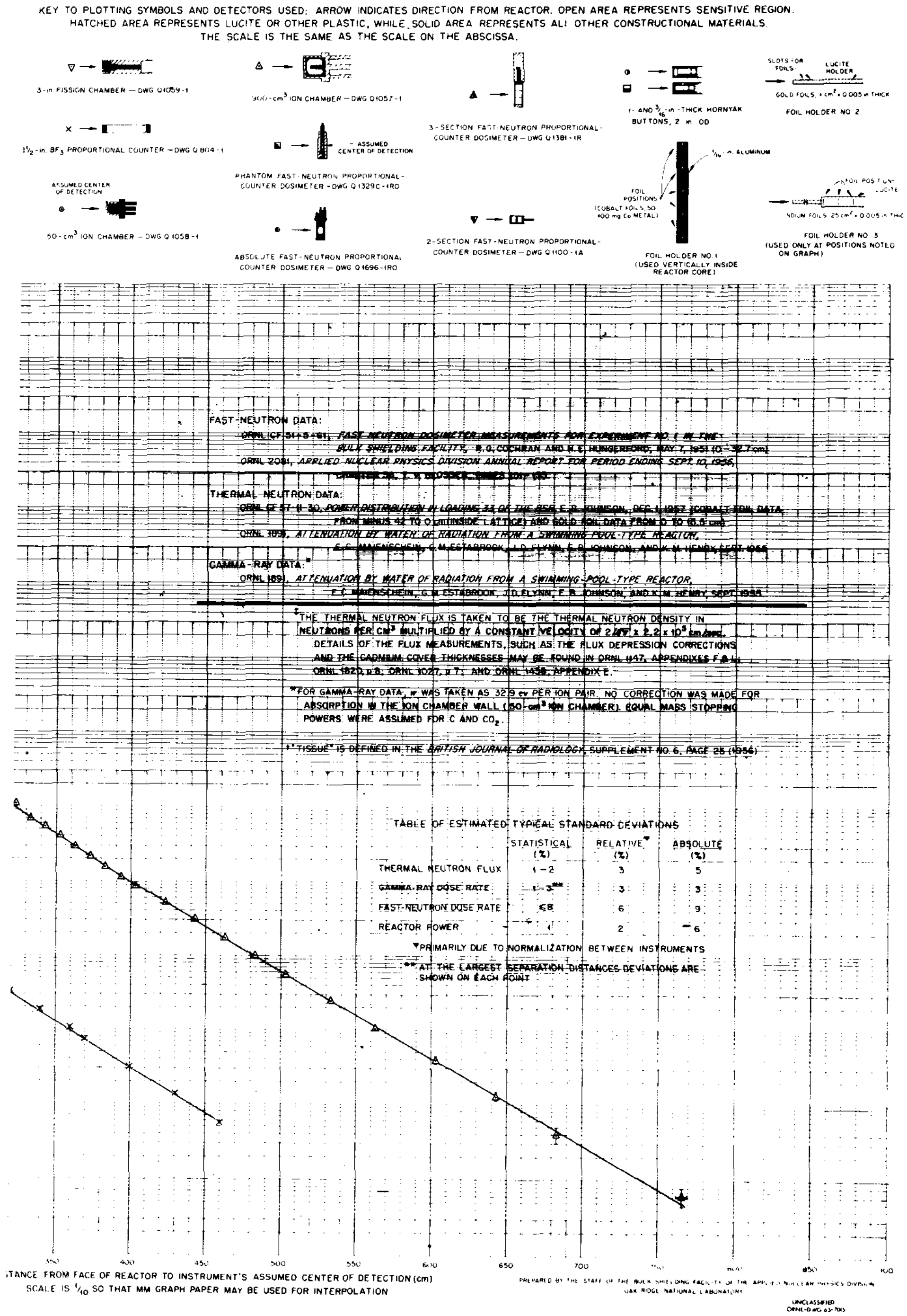

Gamma Dose Rate Through Water.

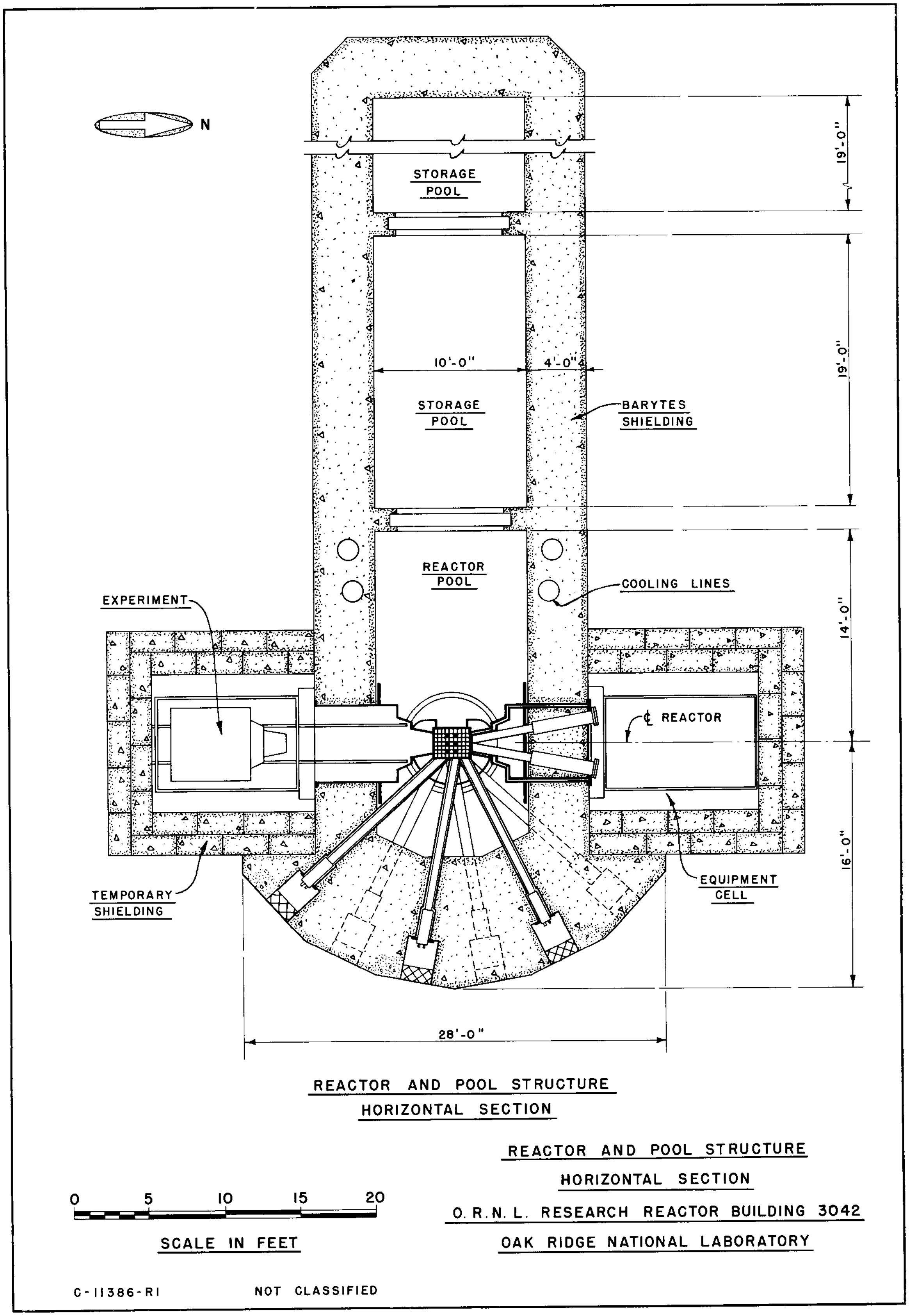

Fig. 9.2. Reactor and Pool Structure Horizontal Section.

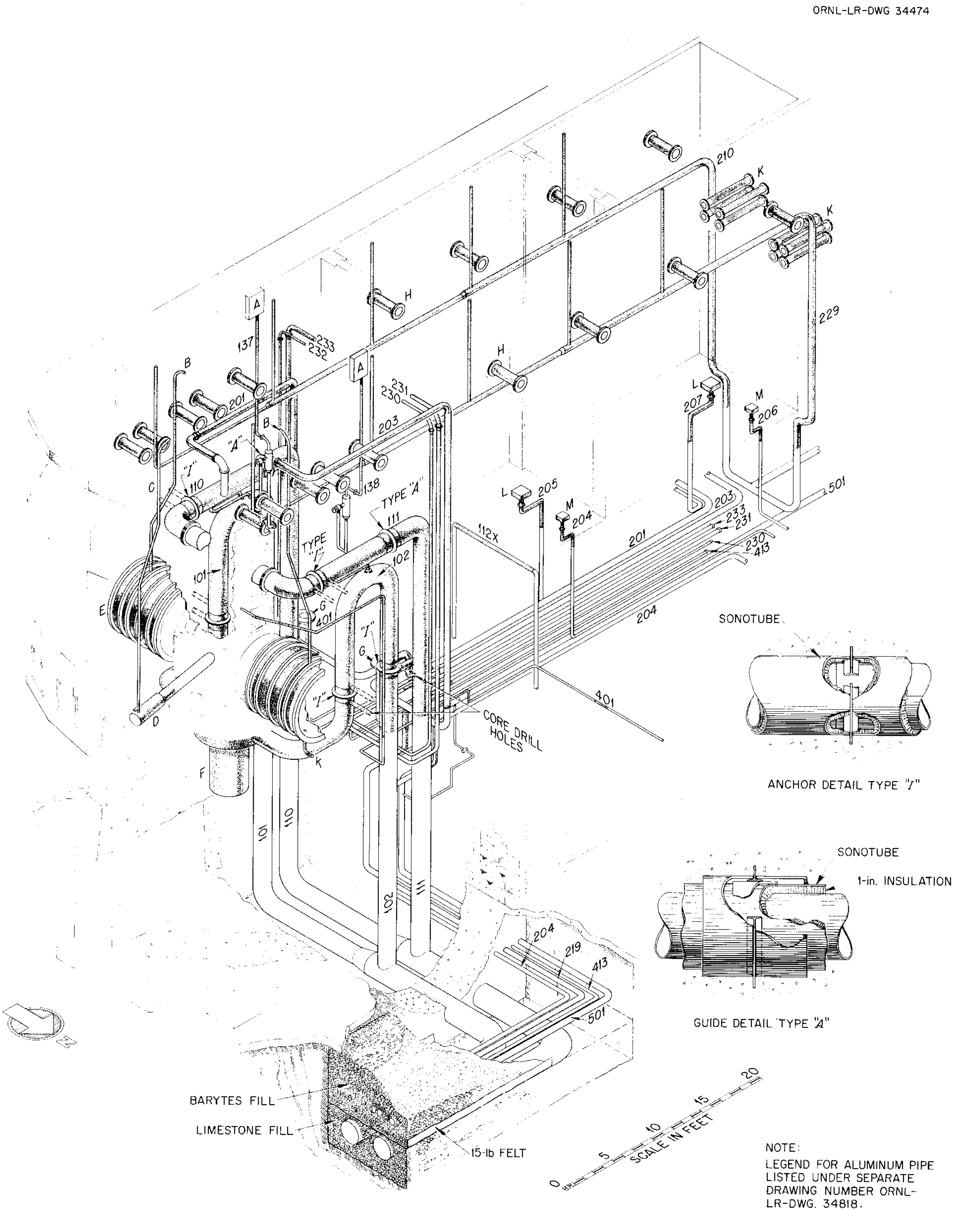

Fig. 9.3. Penetration in Lateral Shield. Stippling on pipes denotes portion embedded in concrete.

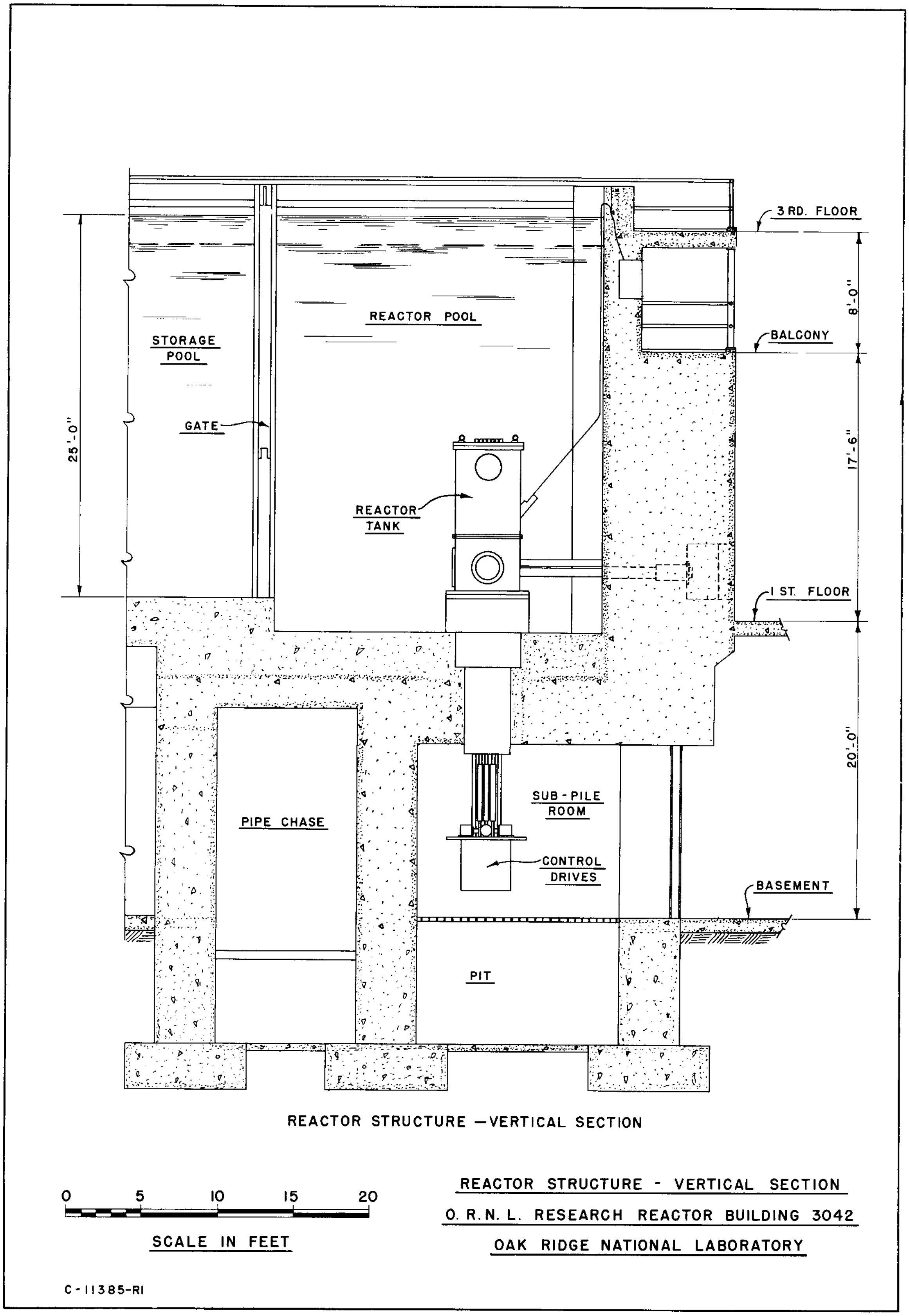

Fig. 9.4. Vertical Section of Reactor Structure.

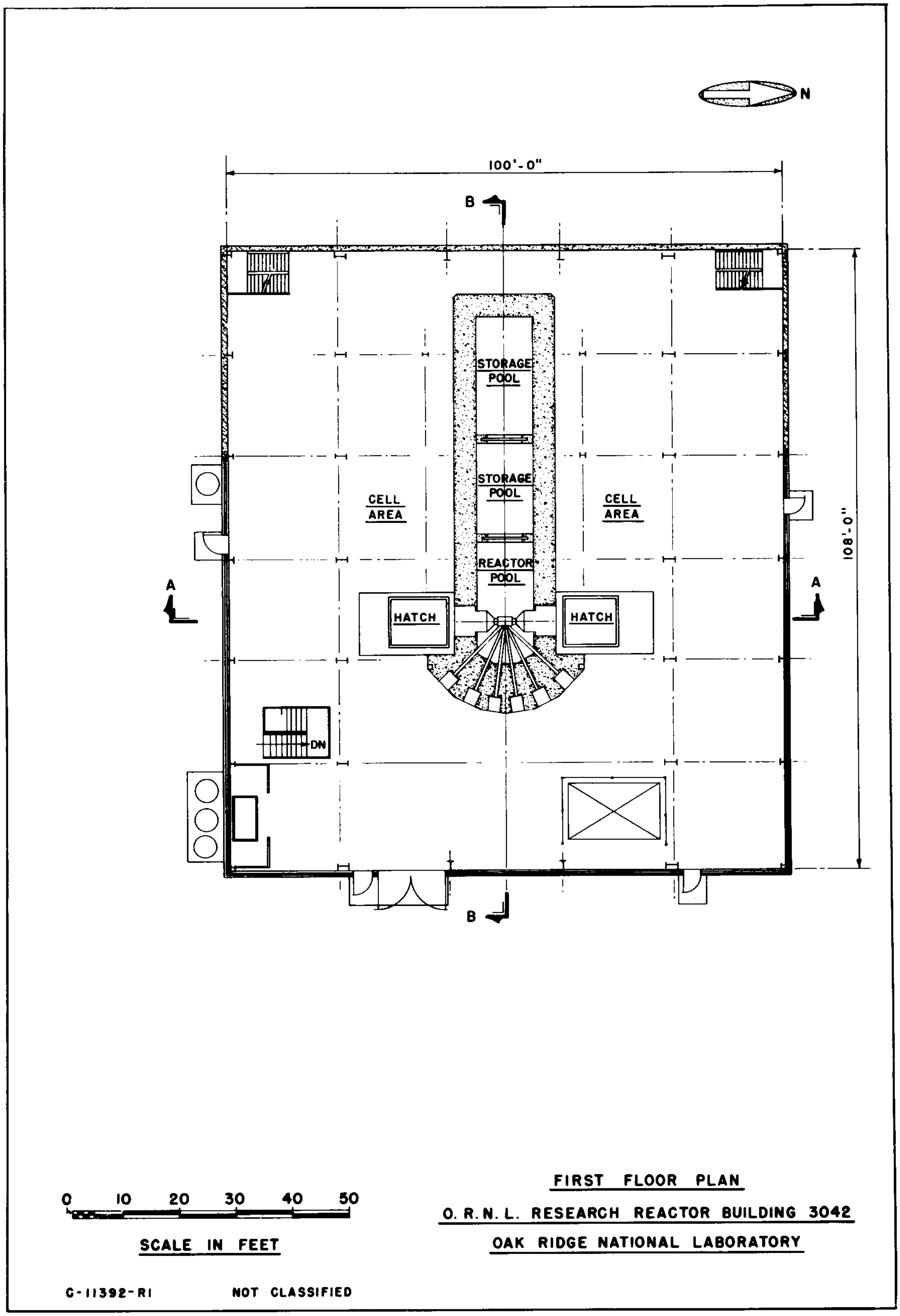

Fig. 9.5. First-Floor Plan.

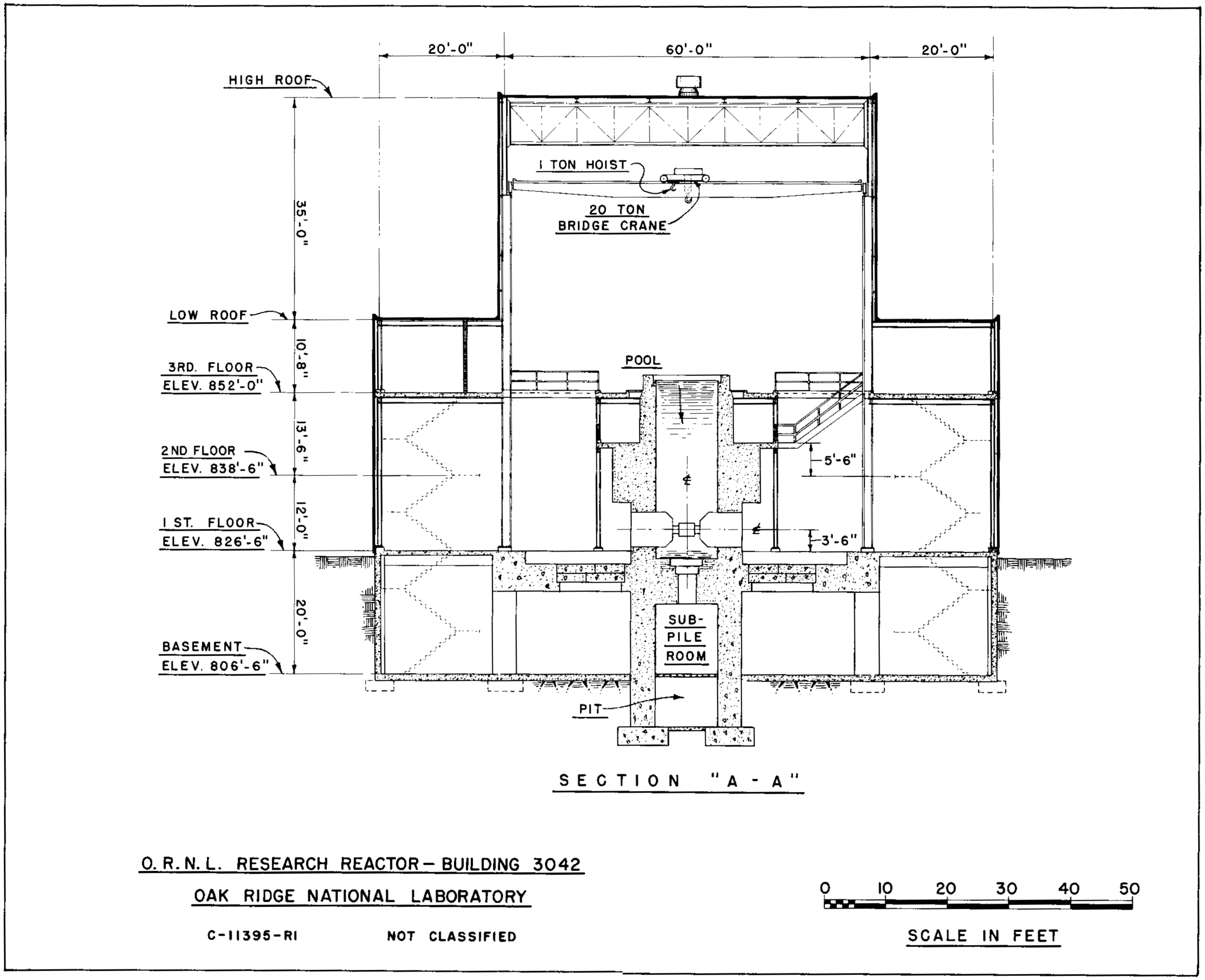

Fig. 9.6. North-South Vertical Section Through Reactor.

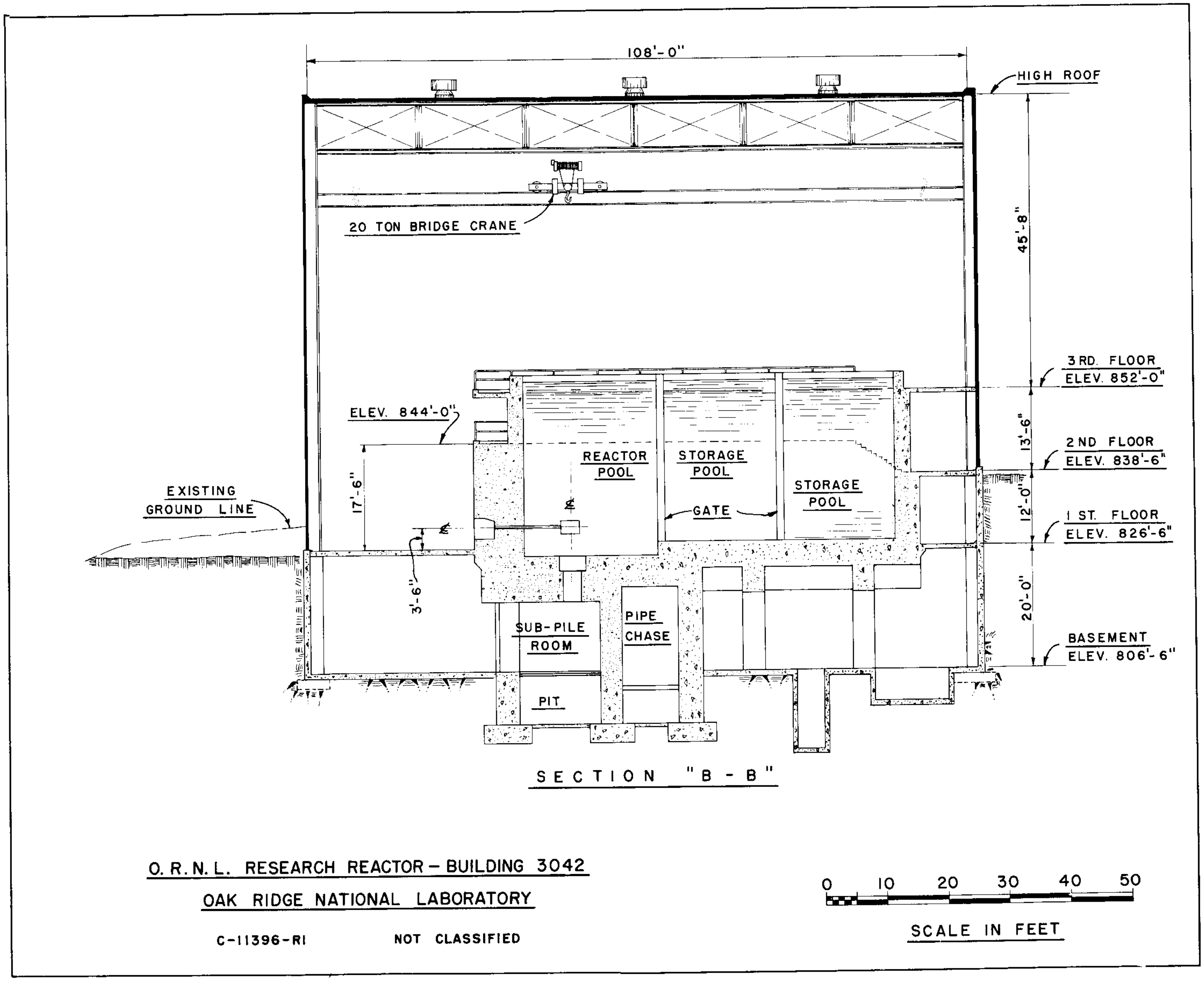

Fig. 9.7. East-West Vertical Section Through Reactor.

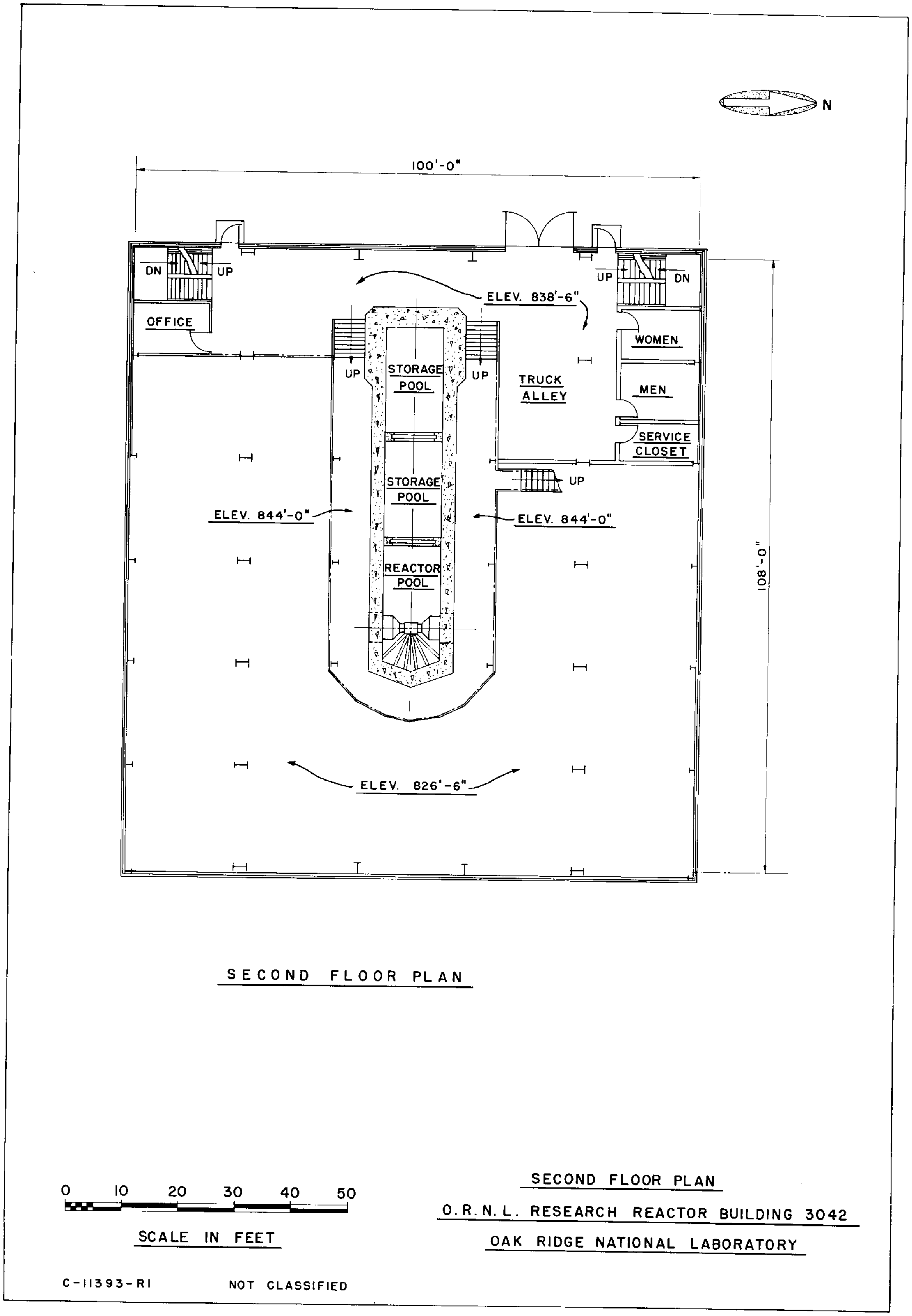

Fig. 9.8. Second-Floor Plan.

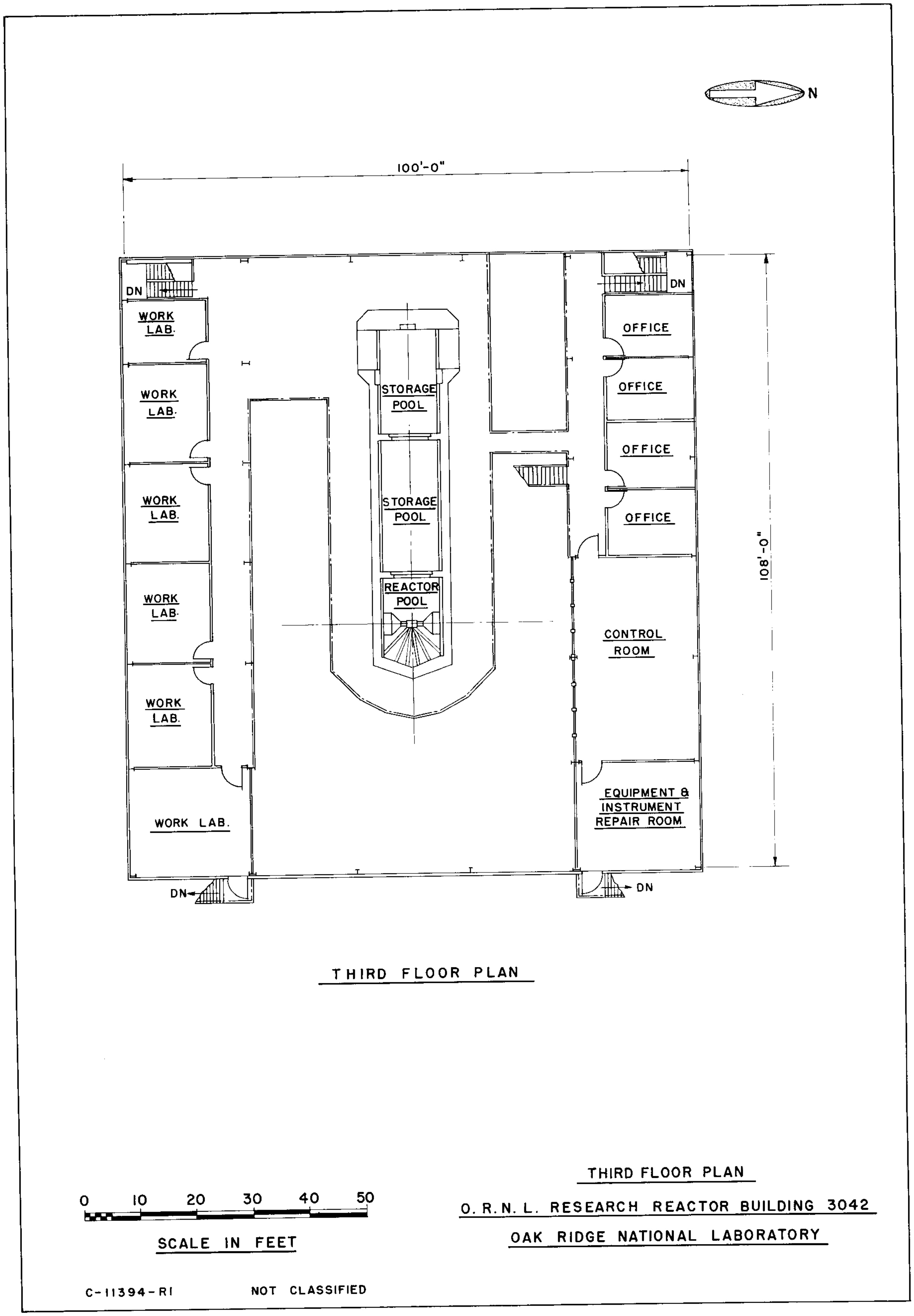

Fig. 9.9. Third-Floor Plan.

Experience indicates that, with the exception of the demineralizer cells, no regions of high radiation intensity (several r/hr) are available to personnel. Access to the demineralizer cells is prevented by locked doors or security gates. It is not necessary to approach this equipment during operation. Lower-level activity (30 to 200 mr/hr) may be encountered in the degasifier call, in the pump house, and in the facility pump cells. These areas are restricted access regions, and local shielding is provided where necessary.

9.2.4 Thermal Shields

A radial temperature gradient is created in the biological shield, particularly in a plane passing horizontally through the core center line, due to the deposition of energy by the attenuation of gamma rays. A stainless steel thermal shield (Fig. 1.3) is installed between the reactor tank and the biological shield where the tank is closest to the pool walls, that is, at the large experimental facilities. The maximum temperature reached in the main shield at 30-Mw operation is insufficient to cause excessive stresses in the concrete. Since the temperature in the interior of the shield is a sensitive function both of the conductivity of the shield and of the heat transfer coefficient at the inner wall, thermocouples were installed in the shield in order to determine the actual temperature distribution and are available for future use if further measurements are necessary.

10. UTILITIES

10.1 Electrical Systems

The ORR is supplied with electricity from the TVA network by the system shown schematically in Fig. 10.1 and in more detail in Fig. 10.2. The TVA network supplies power (161 kv) to ORNL primary substation 0901, where it is stepped down to 13.8 kv. The 13.8-kv bus then supplies substation 4000, which in turn supplies 2.4-kv power to the reactor primary cooling pumps and to substation 14-4, where 480-v power for the reactor cooling-tower fans and the air-conditioning and pool-tower equipment is obtained. Primary substation 0901 also supplies substation 3000, which provides 2.4-kv power for the reactor secondary pumps, and to substation 3-3 for 480-v power for the ORR building. Reduction in the voltage is accomplished, where necessary, by transformers located in various parts of the electrical system.

The components and services which should continue operation during a normal-power outage are furnished electrical power through an emergency-power system. Upon failure of the normal-power system, these components receive power from a diesel-driven generator. The components connected to this system are listed in Fig. 10.2.

Those components whose operation during a power outage is considered of greatest importance are supplied by sets of batteries. Equipment in this category includes the primary-coolant-pump pony motors, automatic electrical switchgear, and certain reactor instruments. These components are supplied with dc power from batteries which are capable of delivering adequate power for at least 2 hr, even in the event of failure of the emergency-power system. The batteries are kept charged by appropriate rectifiers and controls. Individual battery systems are used for each of the three pony motors.

10.1.1 Normal-Power System

The normal-power system furnishes 2.4-kv power to the ORR complex through two independent substations. The loads are assigned to the two substations as shown in Fig. 10.2.

Connections between the 13.8-kv switchgear and the transformers are made with three-conductor paper-insulated, lead-covered, neoprene-jacketed cable, run in underground concrete-encased conduit. The outside transformers are mounted on concrete pads.

Power from the transformers is distributed to the various loads through a number of bus ducts. The distribution of the power sources for the principal loads is shown in Figs. 10.2 and 10.3.

Inside the ORR building, breaker 12 ties power from station 3-3 to the building main power bus. From this bus, power is supplied through breaker 23 to panelboard L-3, thence through circuit 4 to the normal services panel in the control room. Circuits L-1 through L-16 in that panel supply power to rod-drive and fission-chamber motors, control circuits, relays, and annunciators.

Station 3-3 also supplies 240/120-v power to the special services panel in the control room. Circuits L-31 through L-46 in that panel supply power to the sigma and magnet amplifiers, nuclear and process recorders, and nuclear instrumentation. Table 10.1 gives the normal-power distribution and loads in the ORR building.

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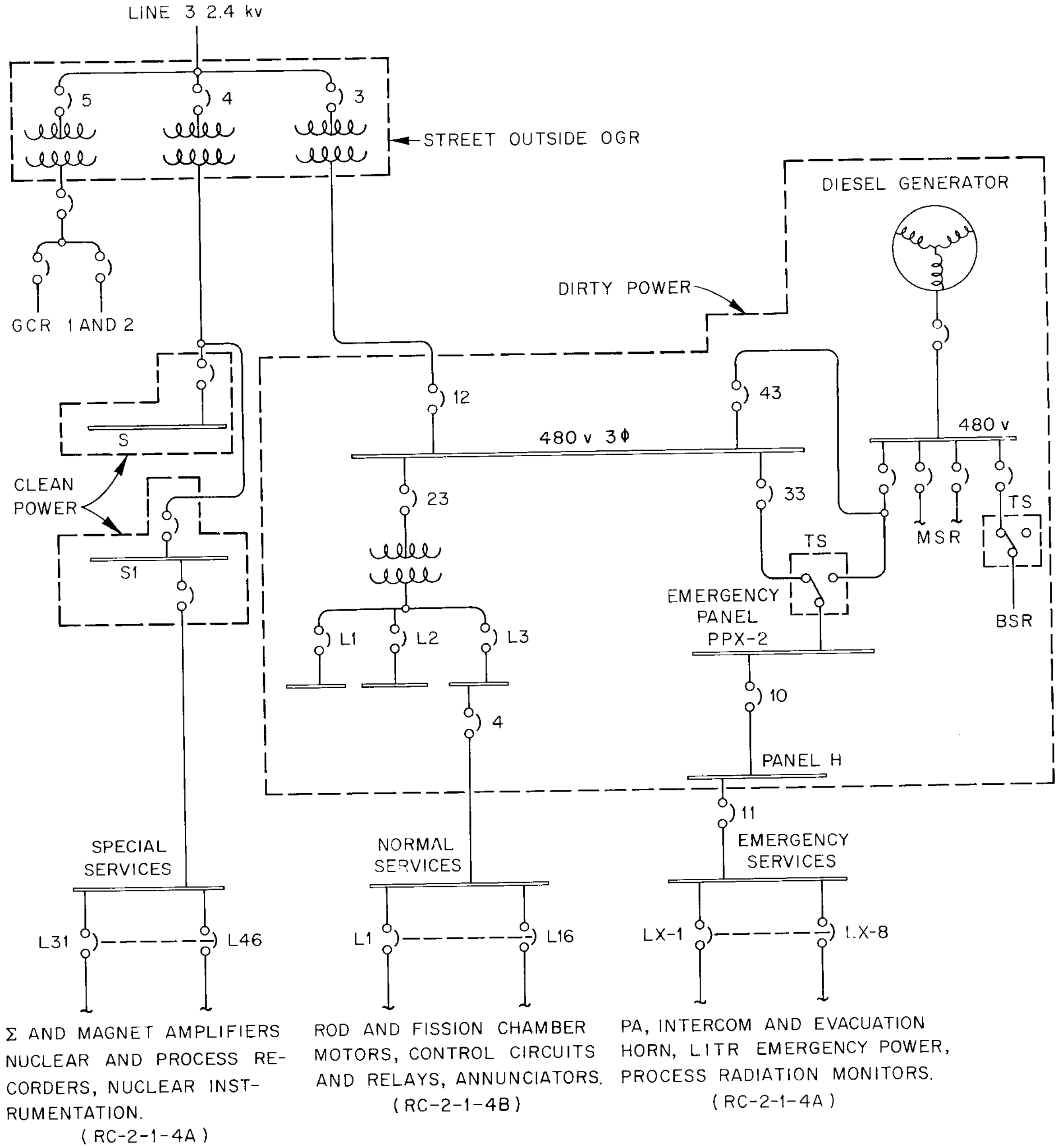

Fig. 10.1. Simplified Power Diagram of the ORR.

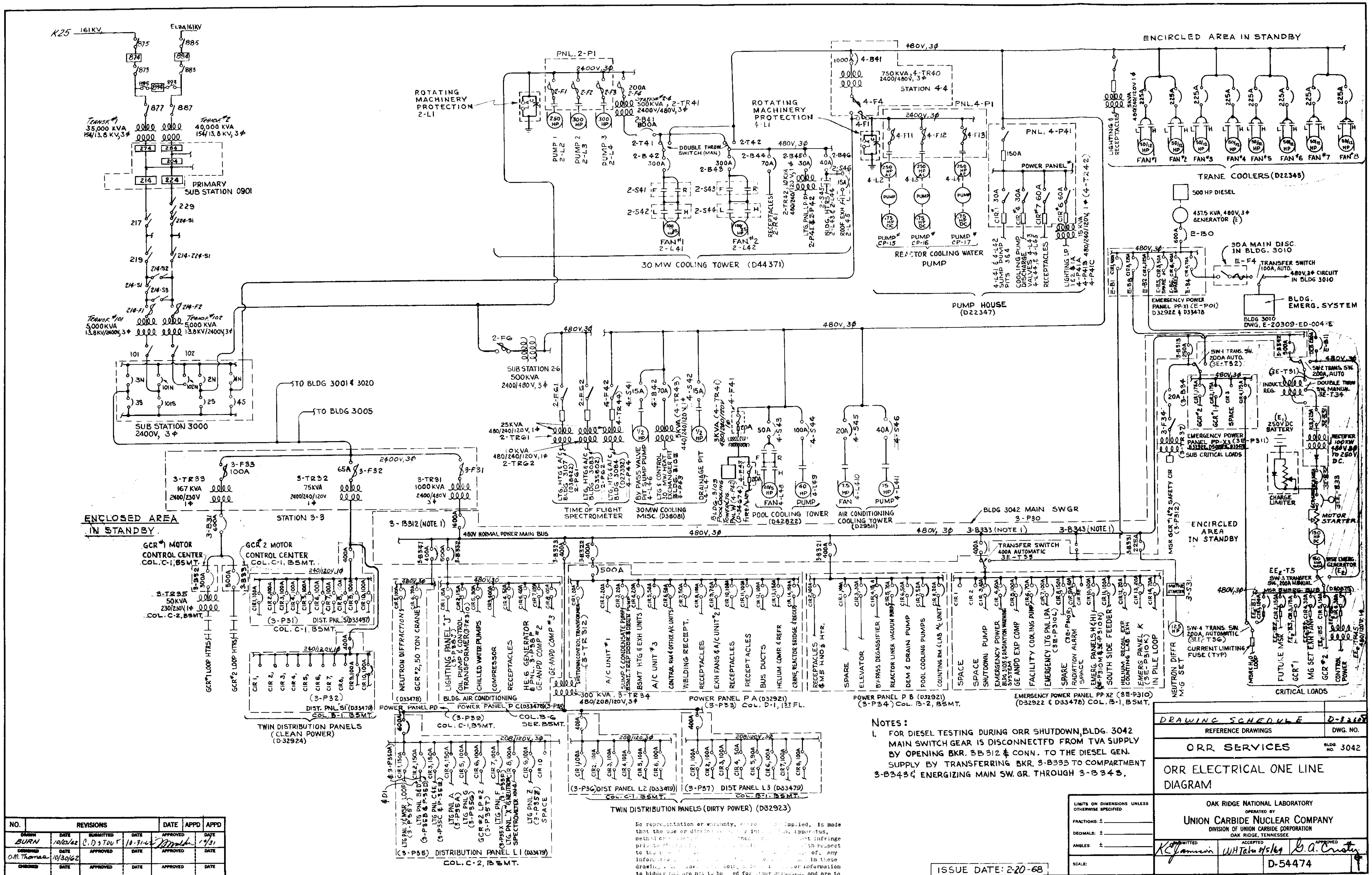

Fig. 10.2. ORR Electrical Diagram.

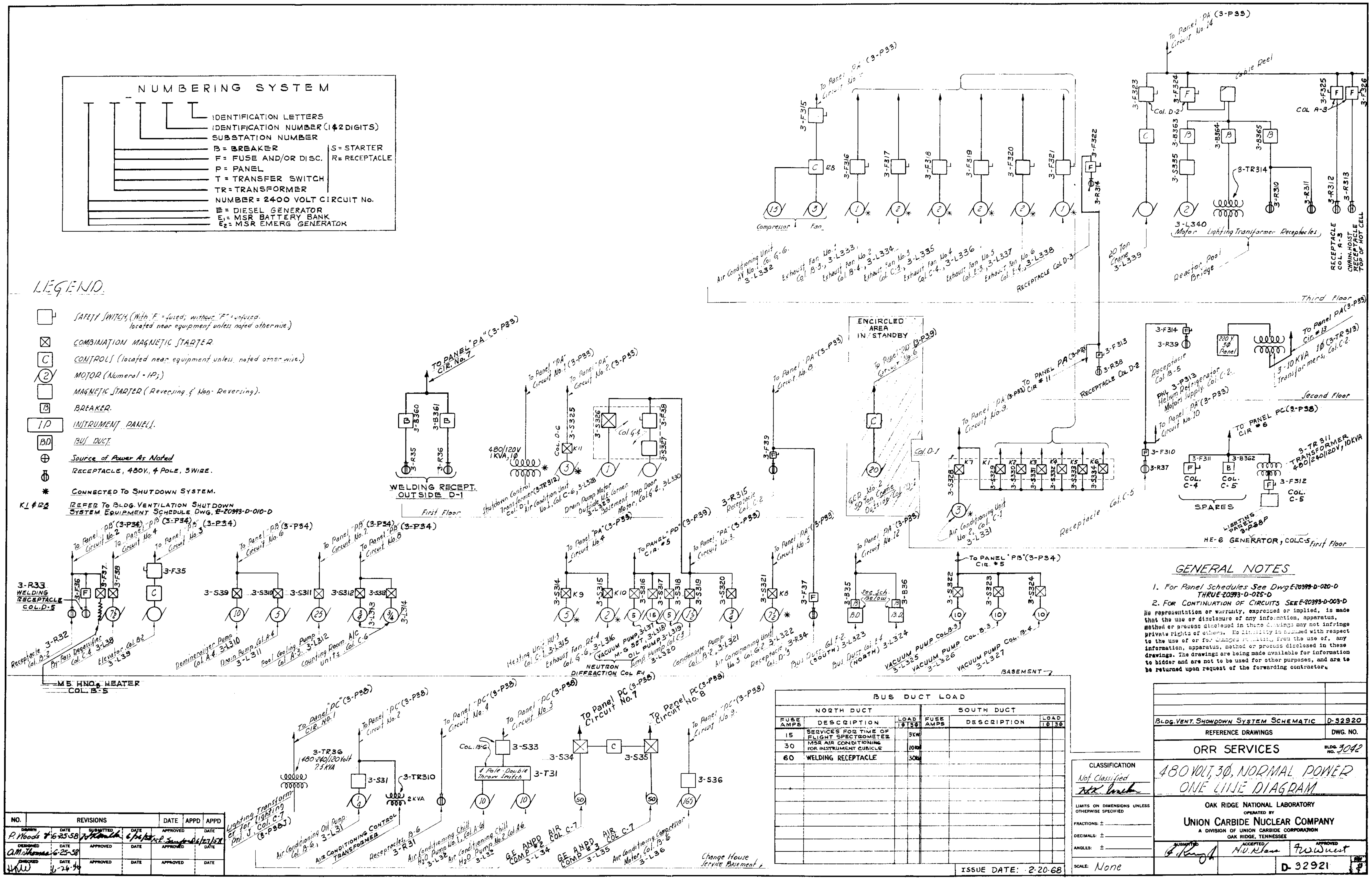

Fig. 10.3. Normal-Power Diagram.

Table 10.1. ORR Electrical Facilities

480-v three-phase normal-power system

Designation	Connected Load (kva)	Usage Factor (%)	Demand (kva)
Building service loads			
Motors (original contract)	92.5	100	92.5
Lighting (total)	200.5	100	200.5
Receptacles (115 v) (total)	32.5	25	8.5
Experimental facilities (twin dist. panels)	220.0	25	55.0
Power receptacles	142.0	25	35.5
Building air-handling apparatus	37.5	100	37.5
Change house			
Air conditioning	176.0	100	176.0
Heating	2.0	100	2.0
Counting room and laboratory	12.0	100	12.0
Second-level laboratories (H_2O Htr)	4.5	100	4.5
Hot cells	5.0	50	2.5
Emergency generator (Htrs)	5.0	100	5.0
Emergency panel H and H1	13.0	100	13.0
Trench drains	1.0	100	1.0
Building 3010 emergency	1.5	100	1.5
Radiation monitoring (in ORR)	12.0	50	6.0
Building 3105 and radiation monitoring	19.0	50	9.5
			662.5
Experiment loads			
Gas-cooled ORR loop No. 1[a,b]	84.5	95	80.0
Irradiation control facility	15.0	100	15.0
In-pile loop N. face (HN-1)	25.0	67	15.0
Fission fragment[b]	135.0	100	135.0
Neutron diffraction	25.0	100	25.0
South side Exp. Fdr.[b]	30.0	50	15.0
Bypass degasifier	7.5	100	7.5
GE-ANPD test facility	134.5	100	134.5
Helium-6 recoil Exp. and HN-4	19.5	100	19.5
Time-of-flight spectrometer	25.0	100	25.0
MSR loop[b]	145.0	54	78.0
Gas-cooled ORR loop No. 2[b]	66.0	93	61.5
			611.0
Less standby loads			274.5
Total experimental loads			337.5
50% overall usage factor			168.5
Building service loads			662.5
Total (all demand)			831.0
Assume 5% overall excess			41.5
Total demand			789.5

[a]Contains 20-kva losses for MSR emergency system.

[b]Loads which are in standby.

10.1.2 Emergency-Power System

The purpose of the emergency-power system is to provide power for certain essential equipment in the event of a normal-power outage. This is accomplished by a diesel-driven generator which is connected in such a way that, upon failure of normal power, the diesel generator will assume the loads in the emergency-power system. The principal components and systems served by the normal and emergency systems are listed in Table 10.2 and are shown in detail in Fig. 10.4.

All emergency-power-system circuits are supplied from the emergency-power switchgear located in the ORR building. During routine operation this switchgear is energized by a 480-v feeder. Although the diesel generator does not run continuously, its main breaker is closed. The diesel starts automatically when a sustained normal-power outage of approximately 10 sec occurs. When the diesel starts, the

Table 10.2. 480-v Three-Phase Emergency-Power (Diesel Generator) System

Designation	Connected Load (kva)	Usage Factor (%)	Demand (kva)
Lighting panel LP-A (3E-P-310L)	5.5	100	5.5
Lighting panel H (3E-P-310M) and H1 (3E-P-310N)	13.0	100	13.0
Shutdown pump	15.0	67	10.0
Gas-cooled ORR loop No. 1[a,b]	84.5	47.5	40.0
HN-1 (in-pile loop)	25.0	67	15.0
Process drain sump pump	5.0	100	5.0
Building 3010 emergency (480 v)	10.0	100	10.0
Building 3005 (FUT)	1.0	100	1.0
South side experiment Fdr.[b]	30.0	50	15.0
Facility cooling pump	30.0	50	15.0
GE-ANPD experiment compressor	34.5	100	34.5
MSR pressurized H_2O loop[b]	145.0	54	78.0
Counting laboratories exhaust hoods	2.5	100	2.5
Gas-cooled ORR loop No. 2[b]	66.0	45	30.0
Radiation monitoring (3E-P-310R)	12.0	50	6.0
Building 3105 and radiation monitoring	19.0	50	9.5
			290.0
System fed from separate 2400-v feeder			10.0
Subtotal			280.0
Less standby loads			163.0
Total demand			117.0
Generator capacity			437.0
Available capacity			320.0

[a]Contains 20-kva losses for MSR emergency system.
[b]Loads which are in standby.

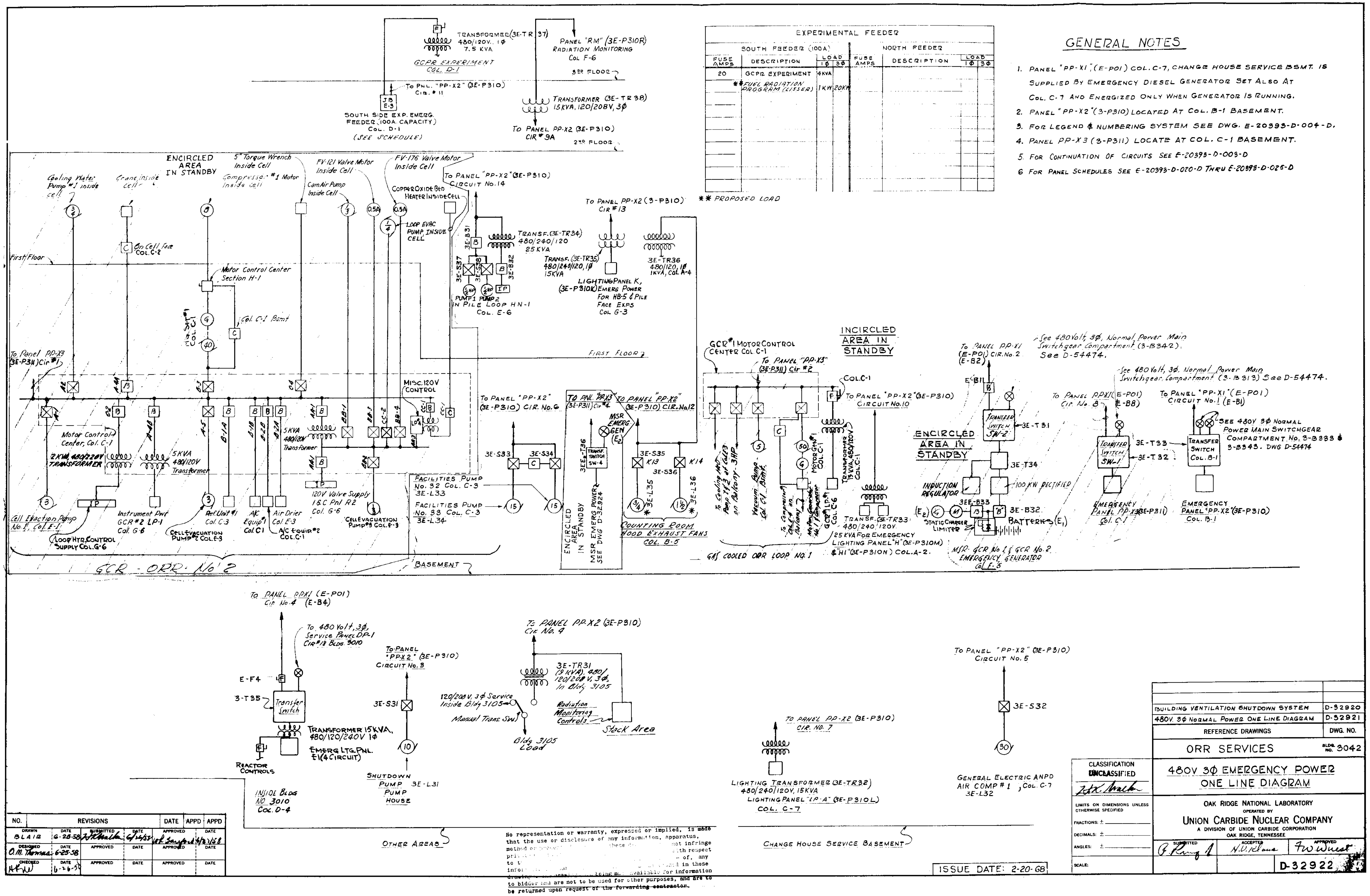

Fig. 10.4. Emergency-Power Diagram.

transfer switch disconnects the TVA supply and connects the generator supply to the emergency-power panel PPX-2.

The diesel is equipped with two battery-powered 32-v dc starting motors whose batteries are kept fully charged by a constant-voltage battery charger.

The four-cycle 12-cylinder diesel engine has a rated capacity of 529 hp at 1200 rpm. The generator is a 1200-rpm, six-pole, three-phase, 60-cps unit rated at 350 kw (438 kw at a power factor of 0.8).

10.2 Plant Water Systems

The ORR plant water is obtained through either of two 24-in. water mains. The normal water main is fed from the ORNL (potable) water reservoir located just north of the ORNL area at an elevation of 1000 ft. The alternate water main is fed from a 3,000,000-gal (potable) water reservoir located on Haw Ridge at an elevation of 1035 ft. Both water mains are connected in such a manner that water is supplied from either or both reservoirs on demand.

The potable water feeds the fire hydrants in a ring main system in the plant. It also supplies drinking water, plumbing needs, secondary-coolant-system makeup, and the cooling requirements of miscellaneous equipment at the ORR.

Supply water to the plant demineralizer system and various experiment requirements is fed from the potable-water system through backflow preventers. Because of these separations, it is convenient to divide the plant water supply into three systems, namely, the potable-water system, the process-water system, and the demineralized-water system.

10.2.1 Potable-Water System

Potable water is supplied from the 8-in. mains of the ORNL potable-water system and is distributed (for the purpose of fire protection) to the reactor building. Water to supply plumbing needs is also distributed from this loop. Water for secondary-coolant makeup is carried in a 6-in. line to the cooling-tower basin, where it is discharged 1 ft above the maximum water level of the cooling-tower basin, thus providing air gap separation.

10.2.2 Process-Water System

The 6-in. process-water line enters the north side of the reactor building from the ORNL process-water system. A flow diagram of the process-water system is shown in Fig. 10.5.

10.2.3 Plant Demineralized-Water System

In addition to the primary-coolant and pool demineralizers, a third independent demineralizer system is provided to produce an adequate supply of water for miscellaneous plant applications.

The demineralizer system consists of two parallel cation-anion units, which allows one pair to be regenerated while the other pair is in service. The system is located in Building 3004, which is approximately 75 ft north of the ORR. The regenerant mixing and distributing facilities in this building are used for sending regenerant solutions to the ORR demineralizers, as well as the Building 3004 demineralizers. A flow diagram of this system is shown in Fig. 10.6.

The Building 3004 demineralizers, when either is in service, combine with a recirculating system to serve a dual purpose. The first includes maintaining the water level in the storage tank between 9 and 11 ft. Although the storage water demand may be small, the demineralizer unit will remain in service at all times in order to be ready for a greater demand. Second, recirculating the water from the storage tank through the demineralizer upgrades the quality of the storage water. It improves the resistivity to a range of 1.5 to 2.0 megohms/cm^3, as compared with a normal makeup single-pass effluent of 0.6 to 0.8 megohm/cm^3. Demineralized water from the Building 3004 storage tank is supplied to various areas of the ORR as shown in Fig. 10.7.

10.2.4 Sprinkler System

Fire protection is provided in the various areas of the ORR complex by one of two types of sprinkler systems. A conventional wet-pipe system is used in the reactor building. In this system, the pipes contain water (under pressure) at all times; fusible plugs in the sprinkler heads will melt and release the water through the heads should a high temperature occur.

Both cooling towers are protected by a dry-pipe system, which prevents freezing of the system during cold weather. The pipes are filled with compressed air which keeps a header water valve closed. Melting of the fusible plug in any sprinkler head releases the air pressure, thereby permitting the valve to open and to allow water to flow through the system to the open sprinkler head.

In all cases, sprinkler heads are separated by not more than 15 ft and are located within $7\frac{1}{2}$ ft of each wall. Hoses for manual use are installed at appropriate locations throughout the ORR.

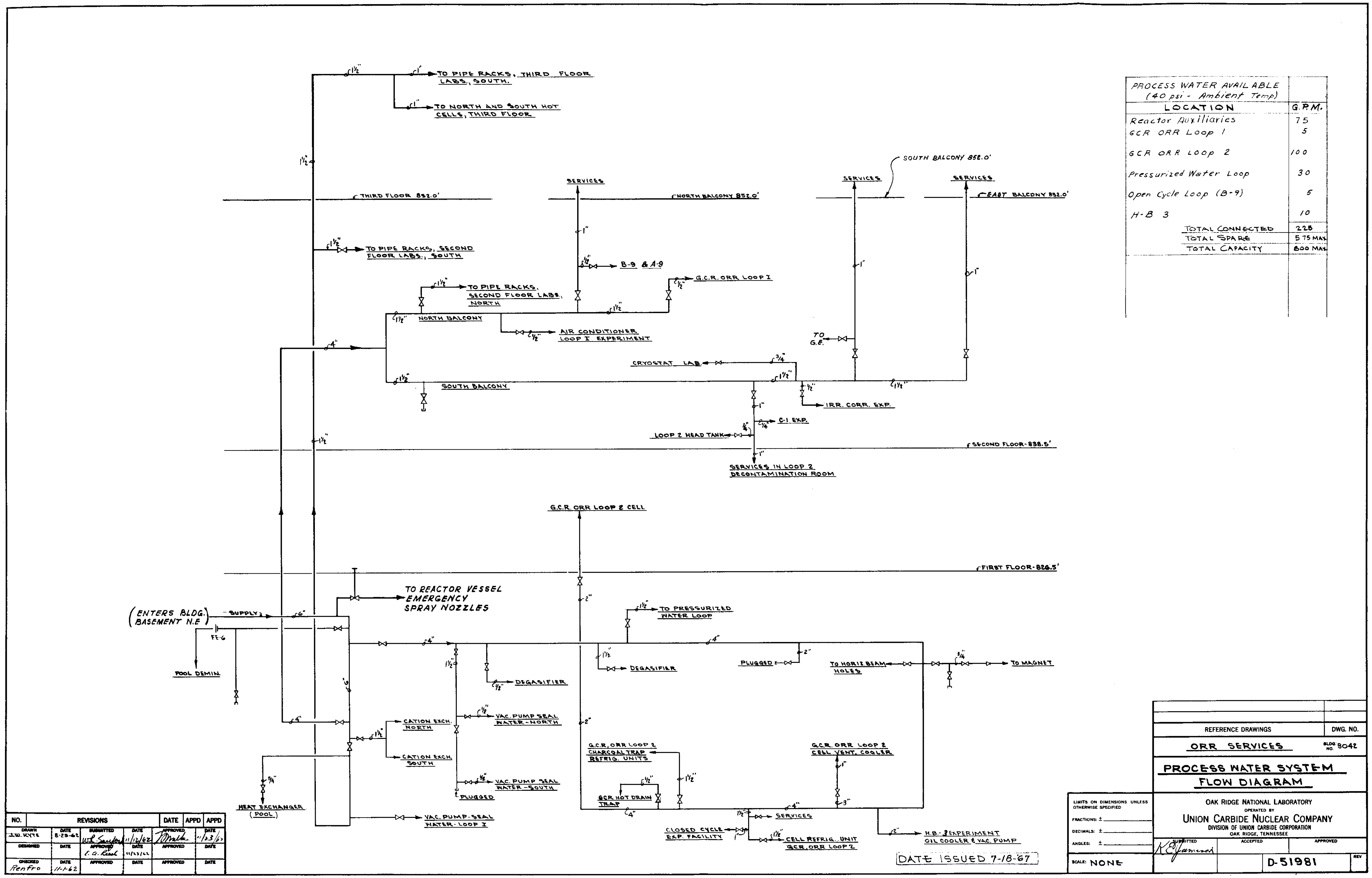

Fig. 10.5. Process-Water System.

ORNL DWG. 65-1910R

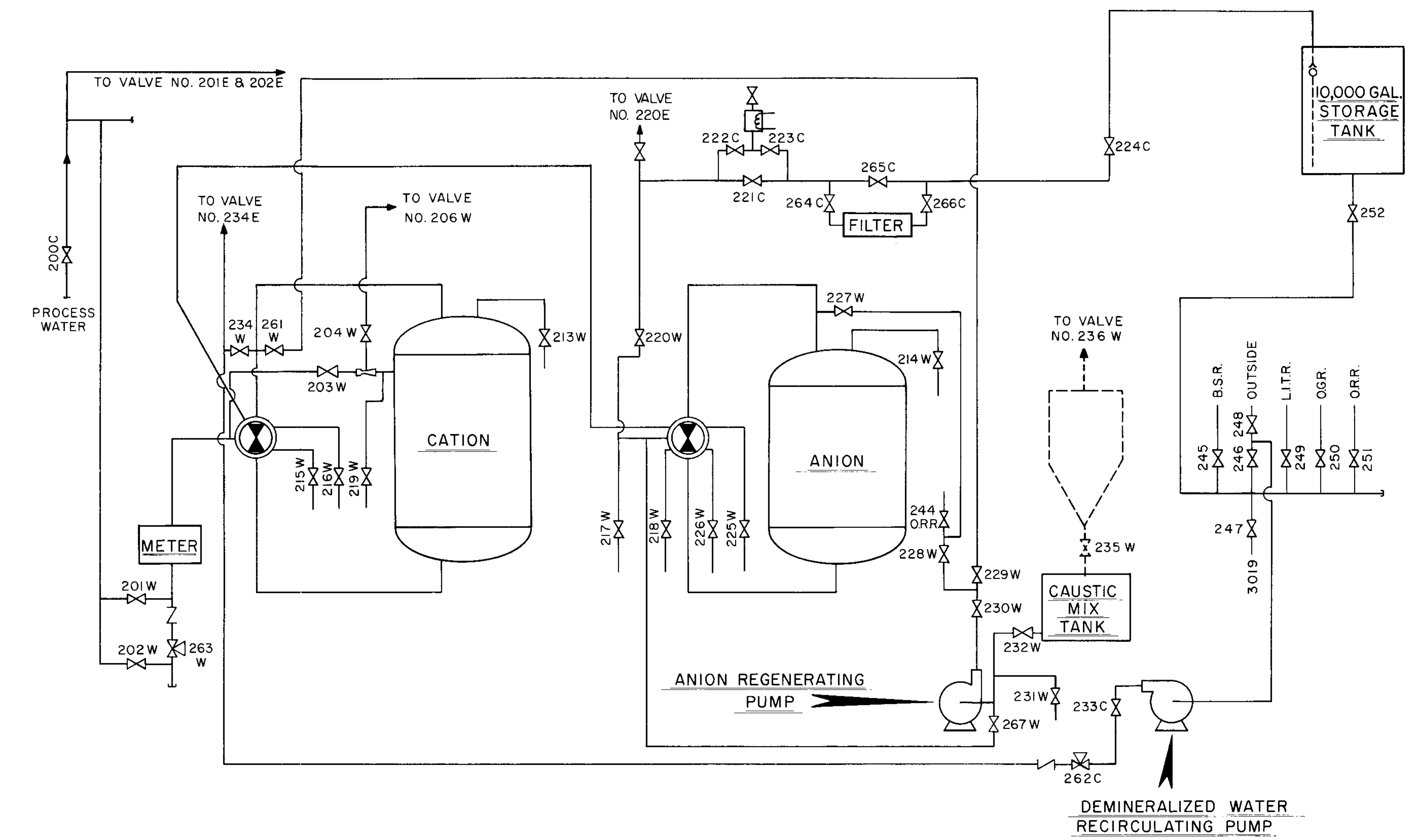

Fig. 10.6. Plant Demineralizer System.

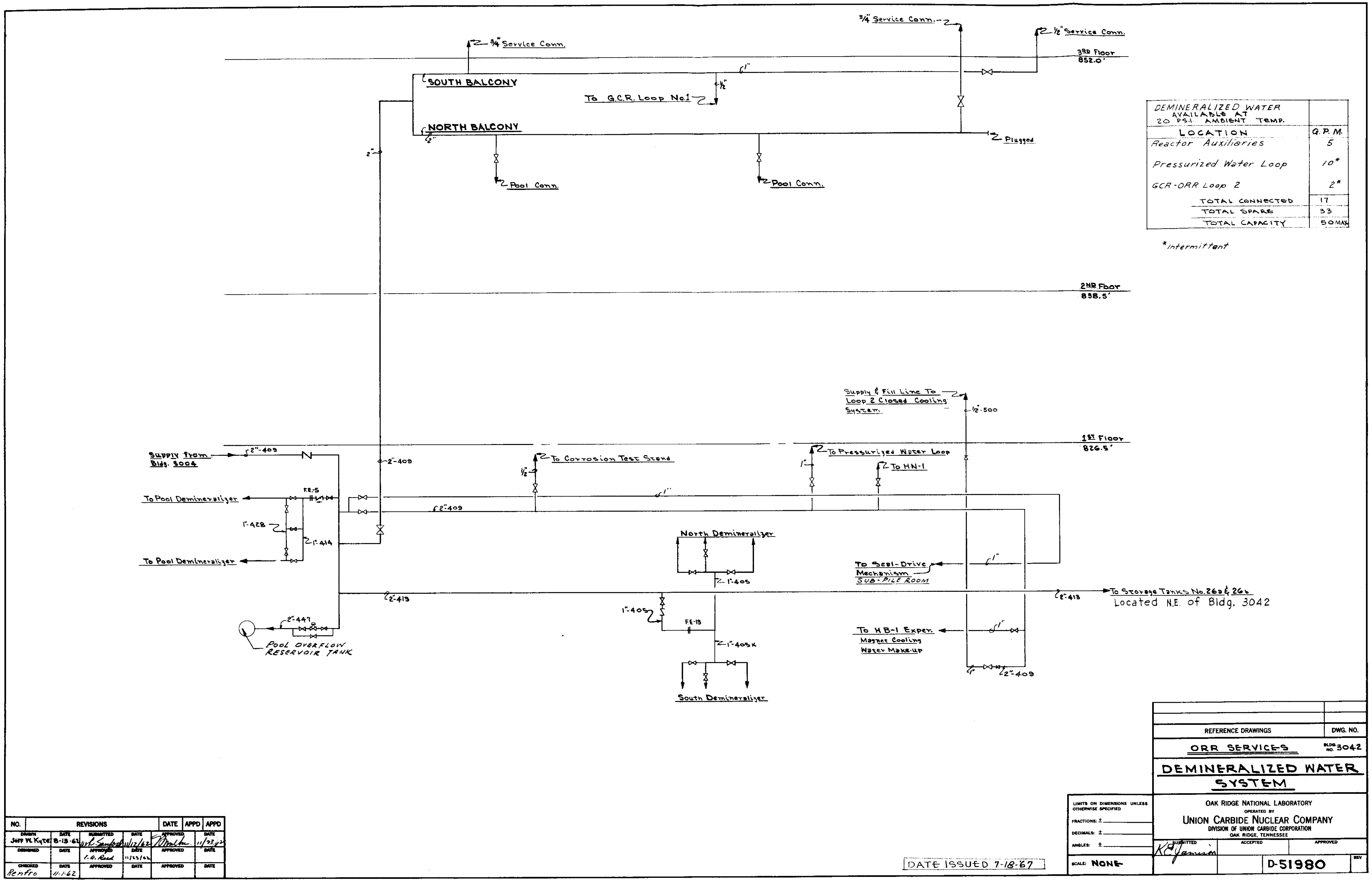

Fig. 10.7. Demineralized-Water System.

10.3 Instrument-Air System

Compressed air for instrument operation is furnished from the ORNL compressed-dry-air system. The air enters the building through a $1\frac{1}{2}$-in. line under 100 psi pressure.

The function of this air is to provide power for the air-operated valves, controllers, and instruments, as well as air to drive hammers, vibrators, or other air-driven equipment of this type.

A flow diagram of the compressed-air system is shown in Fig. 10.8.

10.4 Alarm and Communications Systems

There are six alarm and communications systems at the ORR. These are:

1. area fire-alarm system,
2. area intercom system,
3. sound-powered phone network,
4. dial (Bell system) phones,
5. public-address system,
6. evacuation alarms.

10.4.1 Area Fire-Alarm System

The area fire-alarm system is controlled by four master boxes. Each of these contains the necessary coding relay, which, when actuated by a signal from a temperature-sensitive device or a manual fire box, transmits a coded signal over the ORNL fire-alarm system indicating the location of the fire. All these coded alarms are also given by repeater bells in the ORR area. In addition, fire-alarm horns near the source of the alarm are energized.

Master box No. 1 serves the entire reactor building, master box No. 2 serves the pool cooling tower and the primary pump house, master box No. 3 serves the reactor cooling tower and secondary pump house, and master box No. 4 monitors for malfunction of the reactor-building fire-detection network.

Heat-actuated devices operated either by temperature rate of rise or by high temperature are placed appropriately in all rooms. The maximum separation allowed between detectors is 50 ft. When tripped by excessive heat, these detectors transmit a signal to the appropriate master box, which sounds the coded alarm. The master boxes are powered by batteries located in the central ORNL fire department control center, and the heat-detector circuits are powered by battery and battery-charger systems located in the ORR building.

10.4.2 Area Intercom System

This system allows the control-room operator to page and talk to persons in various locations throughout the area. It also makes it possible to check for unusual noise in some of the equipment areas. The master stations (in the control room, in the Health Physics office, in the secretarial offices, and the supervisors' offices) are capable of calling several stations at once to permit coordination of activity. The electrical power is supplied from the normal and emergency systems.

10.4.3 Sound-Powered Phones

The sound-powered phones are provided primarily for continuous communication between two areas for long periods of time, for example, during equipment checkout. They are also used in high-noise areas and in infrequent-usage areas inappropriate for the intercom system.

10.4.4 Dial Phones

Regular dial phones are located in the offices and at other appropriate stations. The control-room phone has an unlisted number to keep the phone free from unnecessary incoming calls.

10.4.5 Public-Address System

Microphones for the public-address system are located in the reactor control room, at poolside, in the supervisor's office, in the secretaries' offices, and in the maintenance office. Speakers are located in each of the major areas of the building. Additional speakers are located outside to serve the nearby area. Three amplifiers are used in this system: one for inside speakers, one for outside speakers, and a standby. Power for the public-address system is supplied from normal and emergency systems.

10.4.6 Evacuation Alarms

Both local and plant-wide evacuation instructions are given over the local public-address system. A plant-wide evacuation signal comes from the ORNL Emergency Center. The ORR local evacuation alarm, a tone signal obtained from an air-operated horn, is given by the control-room operator. It may be actuated by switches located in the control room and just outside the main personnel entrances to the reactor building.

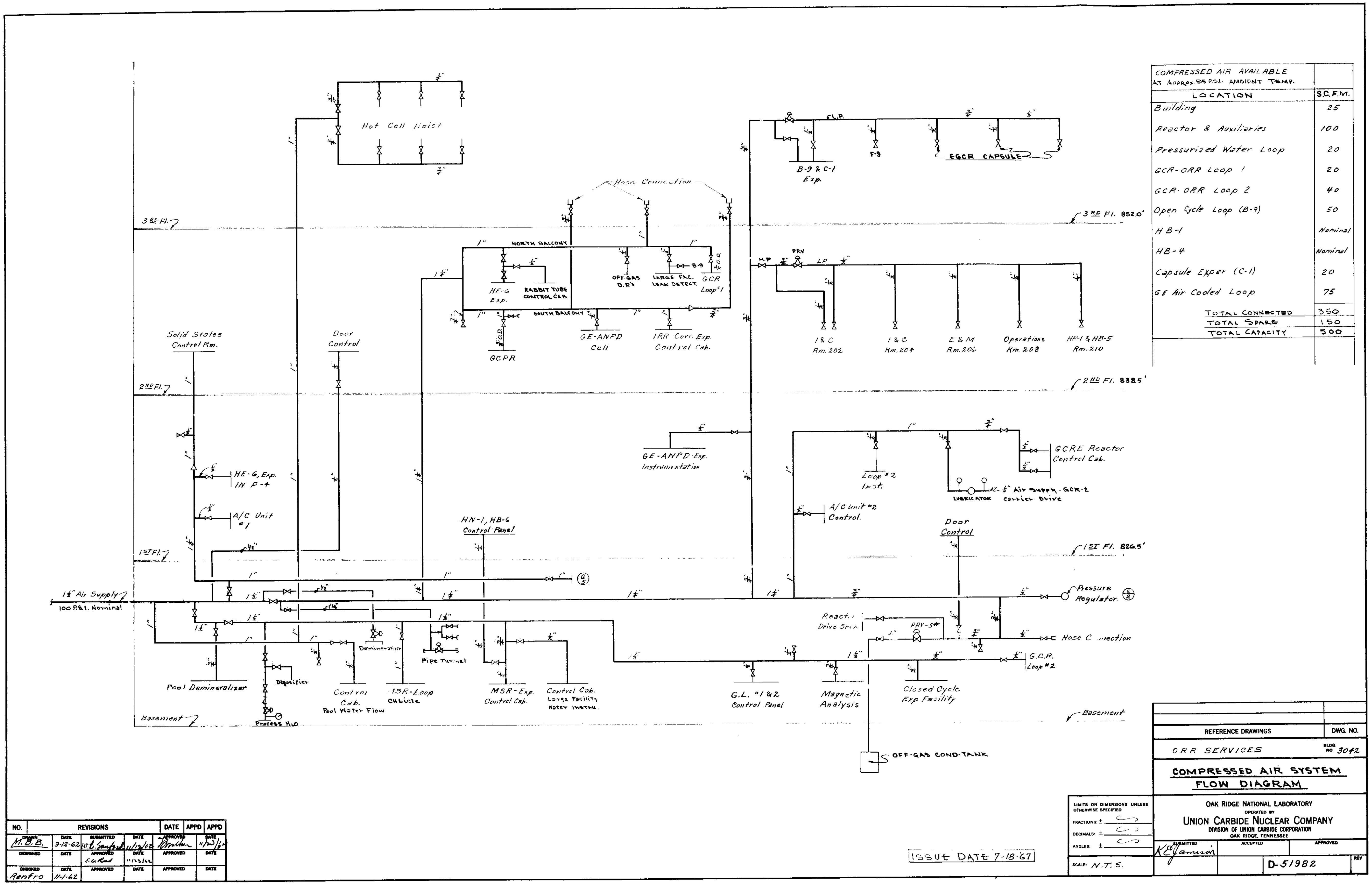

Fig. 10.8. Compressed-Air System.

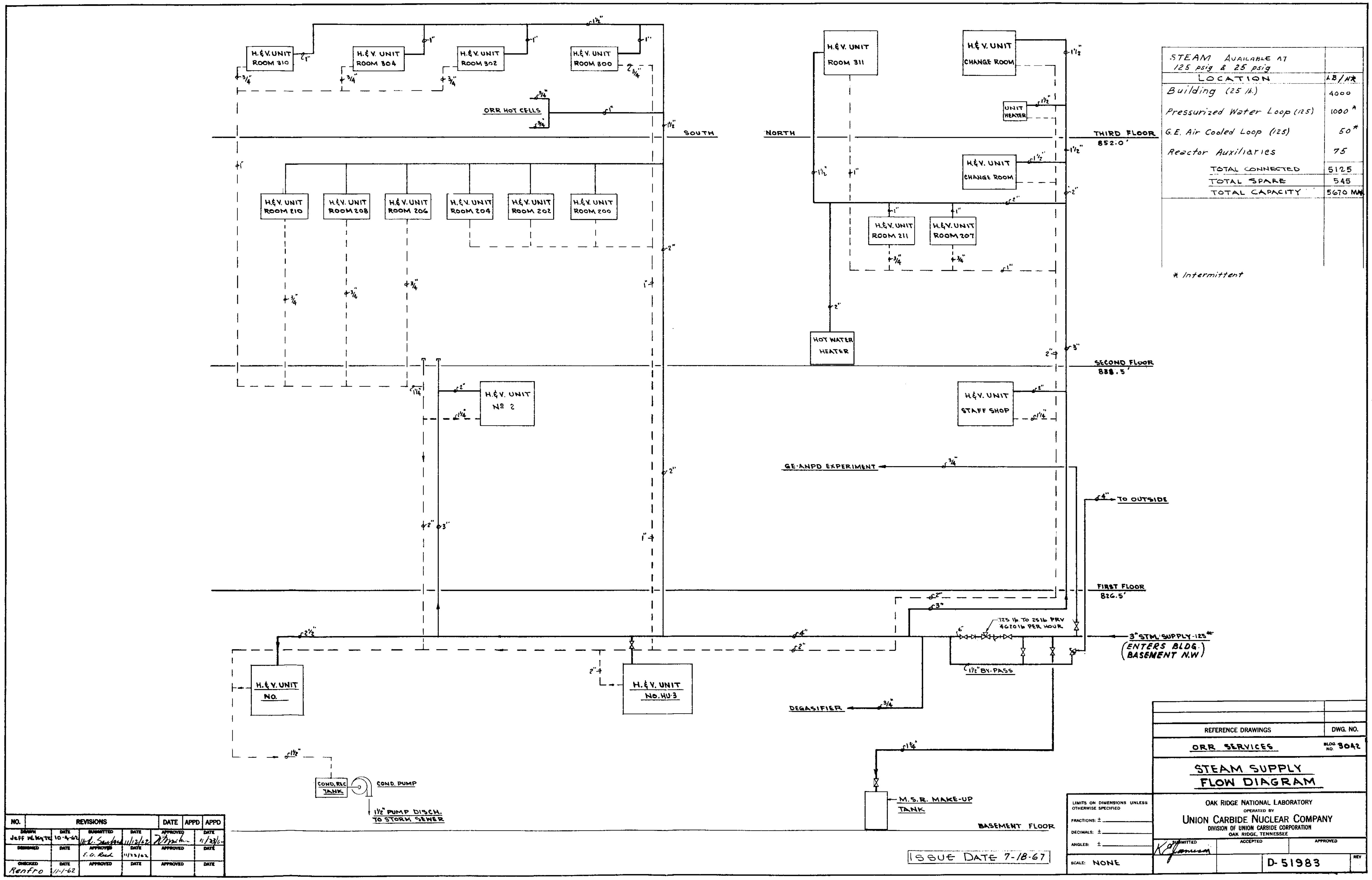

Fig. 10.9. Steam Supply System.

10.5 Steam System

Steam is used for heating the ORR building and for steam-tracing equipment which is subject to freezing during cold weather. The steam is supplied through the ORNL steam system. A flow diagram of the ORR services is shown in Fig. 10.9.

11. SPECIAL SYSTEMS

11.1 Gaseous Waste Disposal

Radioactive gaseous waste is disposed of by the two systems described in detail in Sect. 4: the cell-ventilation system and the off-gas systems. The cell-ventilation system is the exhaust for the reactor dynamic-containment system and is intended to handle infrequent large-volume activity releases, whereas the off-gas systems are designed to dispose of the routine low-volume releases from the various items of equipment. Most of the radioactivity is trapped on absolute or charcoal filters, which are ultimately disposed of by burial, as in the case of other ORNL solid waste. A small amount of activity is discharged to the atmosphere through the ORNL (3039) stack, where it is rendered harmless by dilution, dispersion, and decay.

Nonradioactive gases, generally chemical fumes, are vented directly to the atmosphere. The areas in which these fumes originate are ventilated by the air-conditioning system and roof fans. A combination of pressure control, dampers, and restricting doors is used for this purpose.

11.2 Solid Waste Disposal

Standard ORNL practice is followed in the disposal of solid waste. Nonradioactive solid waste is put into $10 \times 5 \times 5$ ft covered metal containers, which are removed by special trucks. Later the waste is incinerated.

Low-level radioactive wastes (generally sealed in plastic bags) are placed in special yellow $10 \times 5 \times 5$ ft covered metal containers and are removed by truck to the ORNL burial ground. Waste emitting radiation of less than 3.0 mr/hr at the surface may be temporarily stored at the work site in covered yellow garbage cans.

Special procedures are used to remove highly radioactive solid waste. In some cases trucks with shielded cabs are used. For very high levels of radiation, it may be necessary to cut up the radioactive item either under water or in the hot cell and remove the pieces in lead casks. Final disposal is accomplished by burial using ORNL equipment and facilities.[1]

11.3 Aqueous Waste Disposal

All of the aqueous waste from the ORR is, after suitable decontamination, eventually discharged to the Clinch River via one of the small streams flowing through the ORNL area. Laboratory procedures (described in detail elsewhere)[2] ensure that the concentration of radioactive material in the river remains well below the maximum permissible concentration.

Aqueous wastes may be divided into four categories according to the type of treatment given the waste:

1. Storm Sewage. This is untreated nonradioactive waste collected from storm and roof drains or from drains in the administrative areas which are not subject to contamination. This waste is discharged directly to White Oak Creek.
2. Sanitary Sewage. This includes waste from showers, sinks, and toilet facilities. It is sent through the ORNL sewage treatment plant, and the effluent is discharged to White Oak Creek.
3. Process Waste. This originates from various processes which normally produce uncontaminated or only slightly contaminated waste. It is designed to handle aqueous waste having an activity concentration of <10 μc/gal ($\sim$5860 dis min^{-1} ml^{-1}). The waste is treated in the ORNL radioactive waste disposal system and released to White Oak Creek (see Fig. 11.1).
4. Intermediate-Level Waste. This includes primary-coolant leakage, demineralizer regeneration fluids, decontamination and "hot sink" drainage, and all deliberate discharges of radioactive liquids. In general, it includes discharges which have, or are likely to have, activity concentrations in excess of 10 μc/gal. This waste is treated in the ORNL radioactive waste disposal system and released to White Oak Creek (see Fig. 11.2).

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[1]F. N. Browder, *Radioactive Waste Management at ORNL*, ORNL-2601 (Apr. 14, 1959).

[2]J. F. Manneschmidt and E. J. Witkowski, *The Disposal of Radioactive Liquid and Gaseous Waste at ORNL*, ORNL-TM-282 (Aug. 17, 1962).

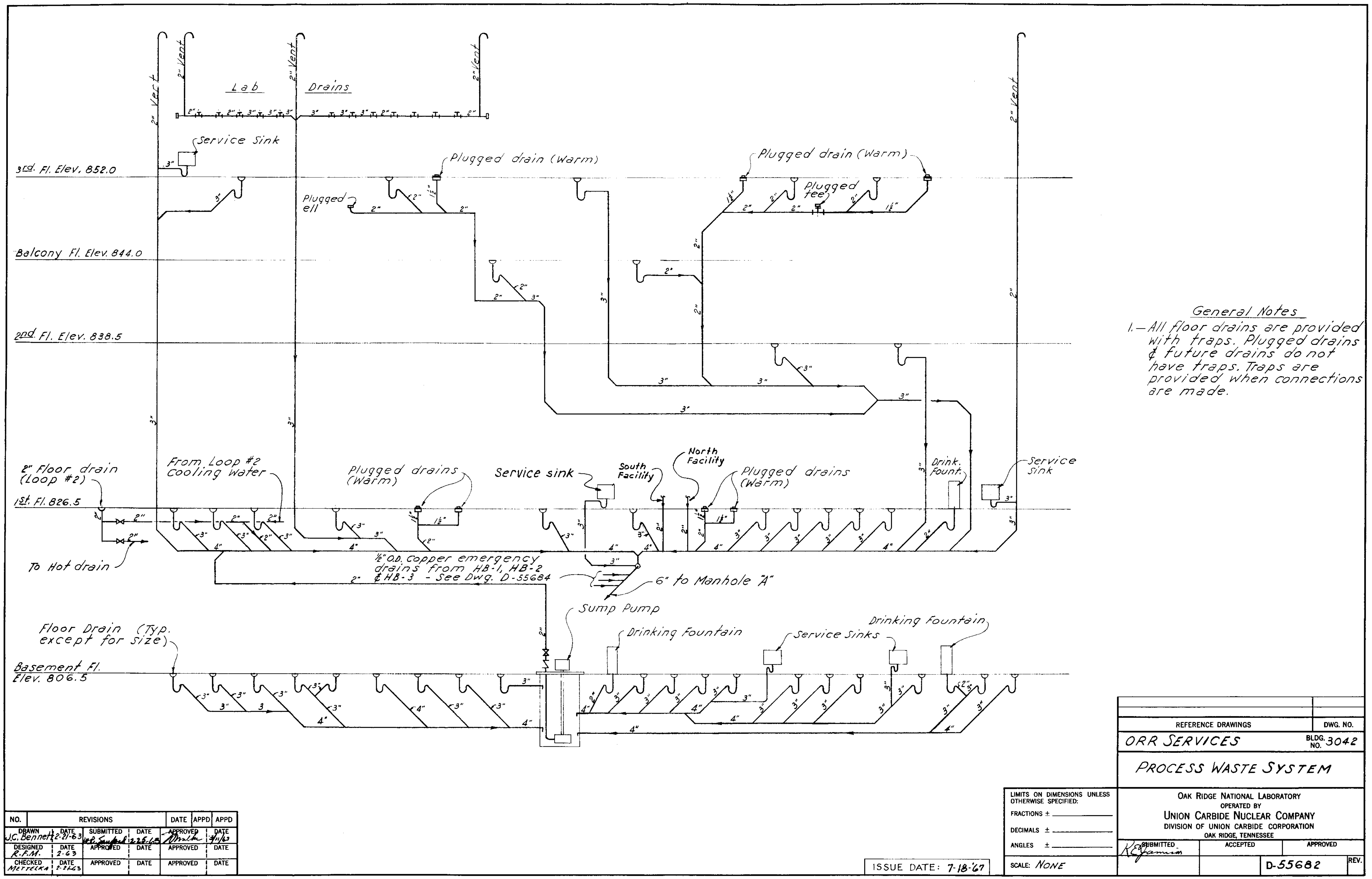

Fig. 11.1. Process Waste System.

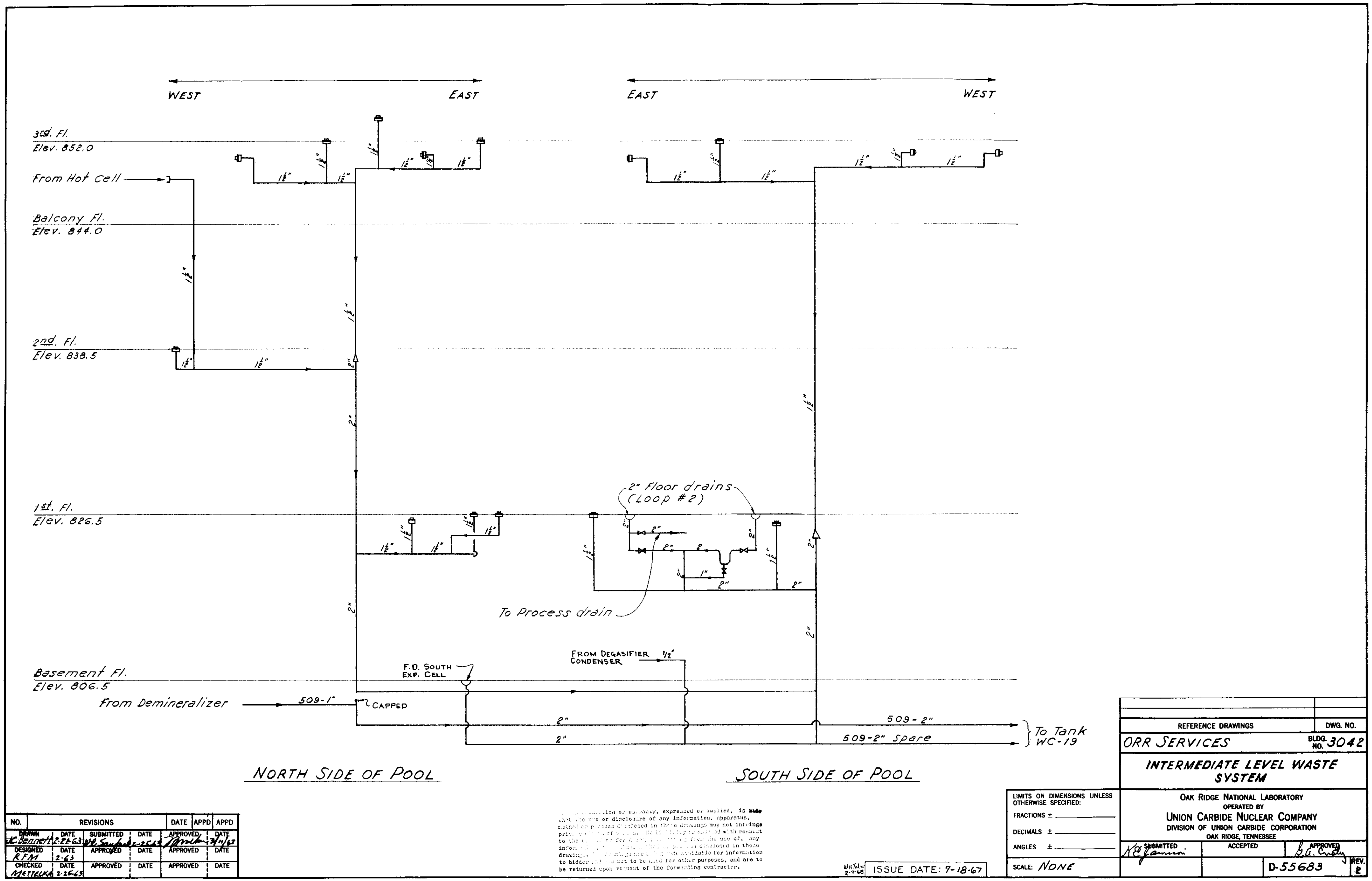

Fig. 11.2. Intermediate-Level Waste System.

12. ORGANIZATION AND ADMINISTRATION

12.1 Introduction

The ORR is under the management of the Operations Division of the Oak Ridge National Laboratory, which is operated for the U.S. Atomic Energy Commission by the Nuclear Division of the Union Carbide Corporation.

The Laboratory is one of the world's largest nuclear research centers, and, since its inception in early 1943, has had a role in virtually every major scientific operation and activity in the atomic energy effort. Today it is engaged in the solution of problems pertaining to almost every aspect of the atomic energy program, particularly those concerned with the peaceful applications of atomic energy. There are nearly 4800 employees in some 100 research groups at work in 8 major fields of interest.

1. Development work in reactor technology has been undertaken on fluid-fuel reactors using both aqueous and nonaqueous fuels, on gas-cooled reactors in the civilian power reactor field, and on reactors for maritime and Army applications. In the past there has been extensive work in the aircraft reactor program.

2. Development work in chemical technology seeks to improve the many chemical separation processes involved in all phases of nuclear energy from ore processing to the purification of man-made transuranium elements.

3. Basic research in the fields of biology, chemistry, physics, metallurgy, and health physics relates to the whole spectrum of atomic energy, but of necessity is focused on problems of current interest in research areas where ORNL is especially well qualified and equipped to work.

4. Specialized training and education is limited to fields in which instruction is not readily available elsewhere, such as reactor operations.

5. Radiation protection through applied biology is directed toward finding better ways and means to detect radiation and radioactive materials, to evaluate the potential hazards they may introduce, and to control radiation and radioactive materials so that people will not be exposed to quantities of radiation which may produce harmful effects.

6. Research and development in the production and use of stable and radioactive isotopes seeks to cut production costs, to increase production methods, and to discover new uses for isotopes.

7. Research into the controlled fusion process and related areas looks toward production of useful power from the fusion reaction.

8. Studies for the Department of the Interior's Office of Saline Water are in progress, involving the chemical properties of water and the technology of materials in aqueous solution.

Nuclear technology studies pursued at ORNL contribute to advances in all areas of reactor technology. These studies include such activities as development and evaluation of conceptual designs; studies of fundamental principles of reactor physics and criticality; and characteristics of reactor shields, instrumentation, fuel elements, alloys, and related items. Of particular significance are the safety-oriented programs, which include management of the Nuclear Safety Information Center, designed to disseminate up-to-date information concerning reactor safety, publication of the *Nuclear Safety* journal, and operation of the Nuclear Safety Pilot Plant. This last program is primarily concerned with the development of engineering information concerning nuclear accidents in order to aid in the safe design of reactor containment and other features.

In addition to the ORR, the Laboratory is currently responsible for the operation of seven other reactors.

The Graphite Reactor was the first installation built at the ORNL site and was operated for 20 years. It produced the first gram quantities of plutonium and later was the principal source of radioisotopes. The reactor operated at 3.5 thermal megawatts and provided space for experimental activities. On November 4, 1963, the Graphite Reactor was removed from active service.

The Low-Intensity Testing Reactor is a natural-water-cooled and- moderated, fully enriched tank-type research reactor. Originally built to serve as a hydraulic mockup of the MTR, it has been operated as a research facility since 1951. The reactor now operates at a power level of 3 Mw.

The Bulk Shielding Facility was designed to facilitate experimental radiation measurements in large-scale mockups of reactor shields. It is now used for other research and experimental purposes. Fueled with enriched uranium, its reactor core is contained in a 40- by 20-ft pool of natural water. It was the first of the swimming-pool-type reactors. The same pool also contains the Pool Critical Assembly, a 10-kw reactor of the BSF design, used primarily for student and operator training and to pretest fuel elements for the ORR.

The High-Flux Isotope Reactor is a beryllium-reflected, light-water-cooled and -moderated aluminum-clad-fuel-plate high-flux-trap-type reactor which uses highly enriched uranium fuel. It first achieved criticality on August 25, 1965, and achieved its design power level of 100 Mw on September 9, 1966.

The Tower Shielding Facility, when in operation, appears as a sphere suspended from cables between four towers that stand 320 ft tall. Inside the sphere is the reactor core, containing uranium-aluminum alloy fuel elements. This unusual facility is used in research on shielding materials, avoiding the confusing effects of reflection from nearby ground and structures. It is fueled with highly enriched ^{235}U and has been in use since 1954, achieving thermal power levels as high as 0.5 Mw. It was originally built for work on problems associated with nuclear aircraft development.

The Molten-Salt Reactor Experiment is being operated to demonstrate the feasibility of long-term operation of the molten-salt fluid-fuel reactor. Extensive development work has been carried out on this molten-salt concept. Fuel for this reactor is a liquid mixture of lithium, beryllium, zirconium, thorium, and uranium fluorides. This program is an outgrowth of work performed at ORNL several years ago on aircraft propulsion. A 10-Mw experimental reactor is in operation at the present time.

The Health Physics Research Reactor provides bursts of radiation for biomedical and health physics research. An unshielded reactor, it is similar to the Godiva reactor at Los Alamos.

Other fields of interest include the following:

Particle accelerators – sources of charged particles for nuclear research – complement the reactor. The Laboratory has both medium-energy cyclotrons and lower-energy Van de Graaff accelerators. A new Tandem Van de Graaff facility, designed to accelerate ions up to 12×10^6 ev, was completed and tested in the spring of 1962. The machine is being used for a variety of experiments in physics and chemistry to study nuclear reactions. The Oak Ridge Isochronous Cyclotron, completed early in 1962, is designed to accelerate positive ions up to 75 Mev and nitrogen ions (heavy particles) up to 100 Mev.

Radioisotopes are produced and packaged at ORNL, a principal center of research in radioactive isotopes and the largest installation of its kind in the world. Currently, more than 1000 shipments of radioisotopes per month are made throughout the United States and abroad. The Laboratory is the site of the Isotopes Development Center, established in 1962 to broaden the technology and application of radioisotopes.

The Health Physics Division of the Laboratory pursues a broad program of research, development, and training in the handling of radiation materials and protection of individuals from radiation hazards. Objectives include the development of improved instruments and better methods for control and disposal of radioactive wastes.

The Biology Division of ORNL carries out the largest biomedical program under Oak Ridge Operations and one of the largest such programs in the entire AEC complex. The experimental projects are directed toward understanding the fundamental changes which occur in living material as a result of the impact of radiation. These include genetic and cytogenetic studies, studies of the basic biochemistry and physiology of cells and tissue, enzymology, radiation pathology, radiation protozoology, bacterial metabolism, cell physiology, plant physiology, nucleic acid enzymology and chemistry, and the chemical basis for radiation protection. Recent studies, undertaken with joint sponsorship of the AEC and the National Institutes of Health, seek to clarify the origins of cancer by attempting to associate chemical effects with those produced by radiation. Associated with this research is the liquid ultracentrifuge being developed with the assistance of the Oak Ridge Gaseous Diffusion Plant to separate viruses which may play a part in causing cancer.

Atomic energy education and training has been an integral part of the activities of the Laboratory since 1943, and part of the present educational program is conducted jointly with the Oak Ridge Associated Universities. This cooperative effort provides an opportunity for university faculty members to engage in advanced research at ORNL and graduate students to complete thesis research toward master's or doctoral degrees at the Laboratory through AEC fellowships and other arrangements.

The Controlled Thermonuclear Program is a major research effort at the Laboratory to harness the power of the fusion reaction, with additional facilities available for study of the basic phenomena of plasmas.

12.2 Organization

The Oak Ridge National Laboratory is organized into 27 line divisions. In addition, there are several staff organizations which have a significant responsibility with respect to reactor operations. A general organization chart of the Laboratory is shown in Fig. 12.1.

Operation of the ORR is the direct responsibility of the Operations Division, which is also responsible for the High-Flux Isotope Reactor (HFIR), the Low-Intensity Testing Reactor (LITR), the Bulk Shielding Facility reactor (BSR), and the Oak Ridge Graphite Reactor (OGR), now shut down. The Operations Division itself is organized into three line and two staff departments, as shown in Fig. 12.2.

In addition to the Operations Division personnel, who perform both day-to-day operation of the four reactors and provide technical assistance, the services of members of three other divisions are required. These are the Plant and Equipment Division, which supplies mechanical maintenance services; the Instrumentation and Controls Division, which not only is responsible for the routine maintenance of the control and process instrumentation but also participates actively in the development of improved equipment and procedures; and the Health Physics Division, which furnishes radiation monitoring service on a routine basis. In each case a technician or group of technicians and a supervisor are assigned to the Operations Division. These service department supervisors work in close cooperation with the superintendent of the Reactor Operations Department and the various reactor supervisors who are directly responsible for operations. In practice a small highly trained group is always available; when needed, additional personnel are made available from the parent organizations.

The organization is shown in Fig. 12.3. The normal paths of communication with the ORR groups are indicated by dotted lines.

The personnel[1] of the Operations Division may be divided broadly into three categories:

[1]F. T. Binford, "Training Programs for Reactor Operations and Reactor Hazards Evaluation at ORNL," in *Proceedings of Symposium on the Programming and Utilization of Research Reactors, Vienna, Austria*, Academic, New York, 1962.

The nuclear reactor engineer possesses a high degree of technical competence and knowledge, including the ability to analyze and treat the various aspects of reactor technology. The senior reactor supervision and technical support personnel of the Division fall into this category. The nuclear reactor engineer has a thorough working knowledge of the physical principles associated with the design and operation of a nuclear reactor and its ancillary facilities. His educational background includes an undergraduate degree in engineering or one of the physical sciences plus considerable specialized work in reactor technology.

The operations engineer is distinguished from the nuclear engineer in that the former is generally charged with the direct responsibility of supervising the implementation of procedures established by the latter. The shift engineers, included in this classification, usually are required to have an undergraduate degree in engineering and some specialized training, often acquired on the job, in nuclear engineering. The training of the operations engineer emphasizes familiarity with the reactor and other devices under his control and their behavior and, in particular, their limitations.

The reactor operators are the individuals who, under the supervision of the reactor engineer or the operations engineer, perform the actual manipulations required to operate the facility. In general, these persons are trained on the job to perform essentially repetitive tasks. They are required to be emotionally stable and manually dextrous and to have a reasonable degree of intelligence. The educational requirement is a high-school diploma or equivalent.

12.3 Training and Qualification of Reactor Operations Department Personnel

All of the engineers currently employed by the Division have undergraduate degrees in engineering or in one of the physical sciences, or equivalent, and they have an average of five years of reactor operating experience. Thus there is a reservoir of trained supervisory personnel which is used for the operation of the ORR.

New personnel hired for the Reactor Operations Department are subjected to an intensive course of instruction. This program consists of a series of on-the-job sessions in which the trainee is taught by an experienced engineer to do each of the tasks

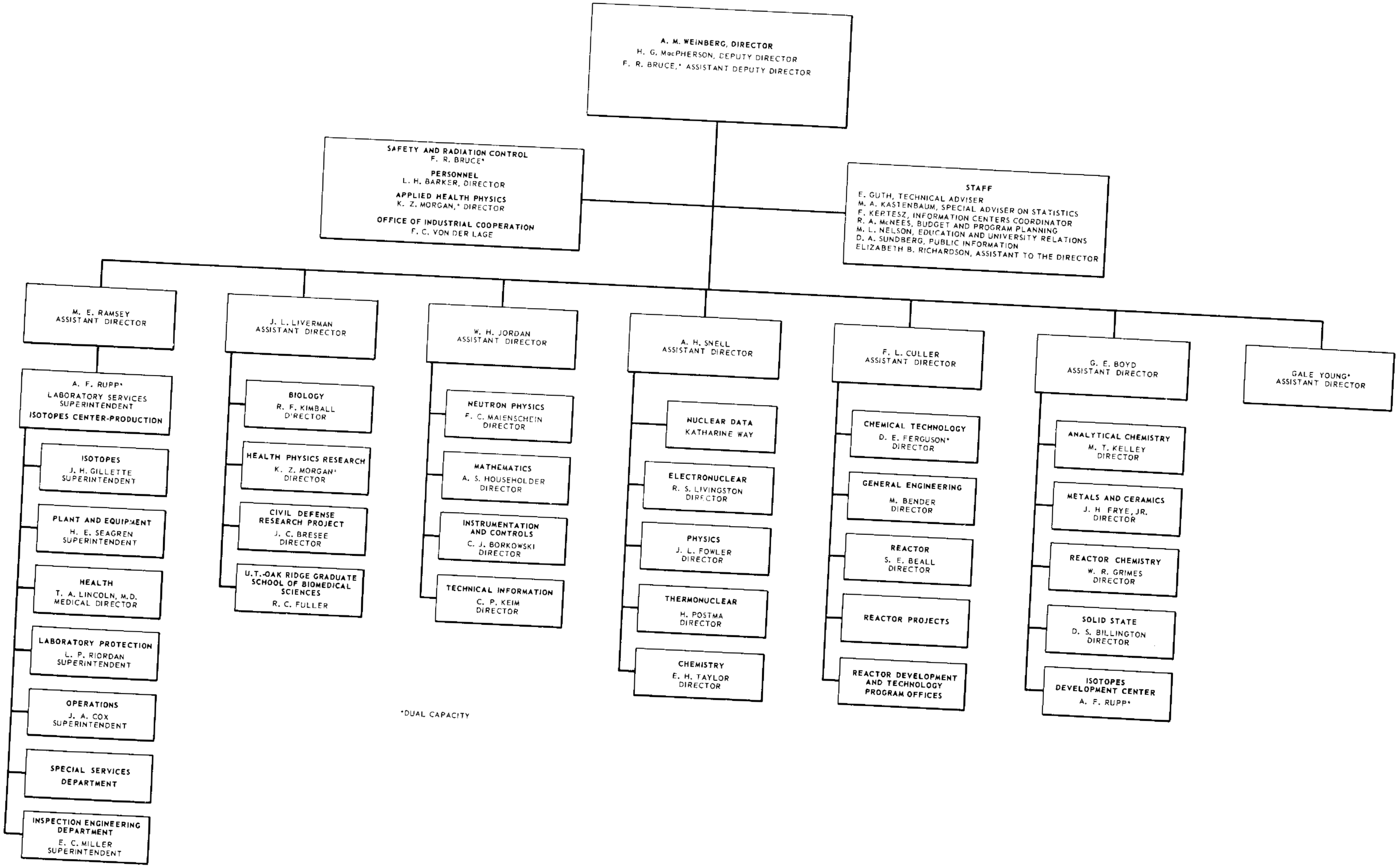

Fig. 12.1. ORNL Organization Chart.

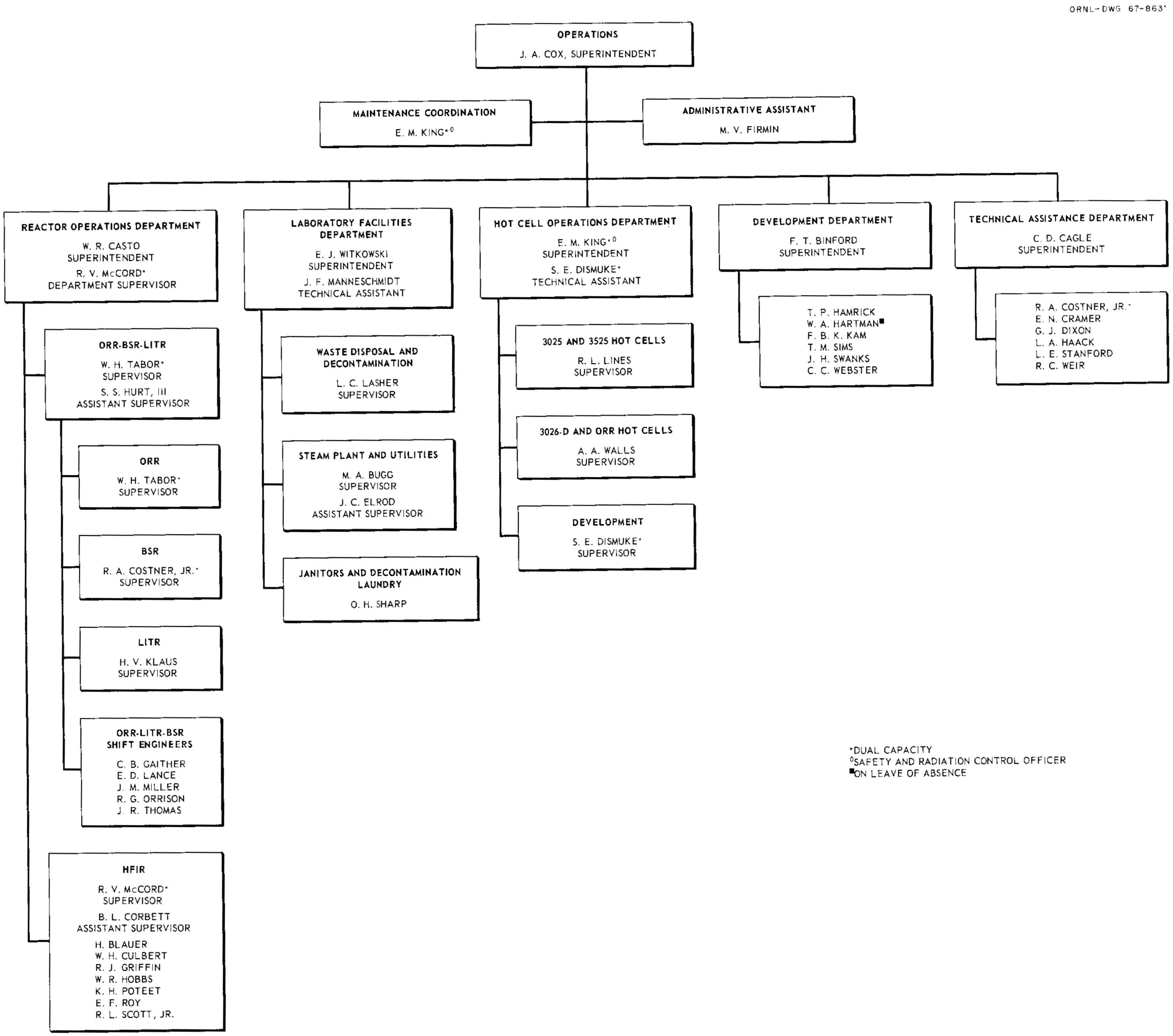

Fig. 12.2. Operations Division Organization Chart.

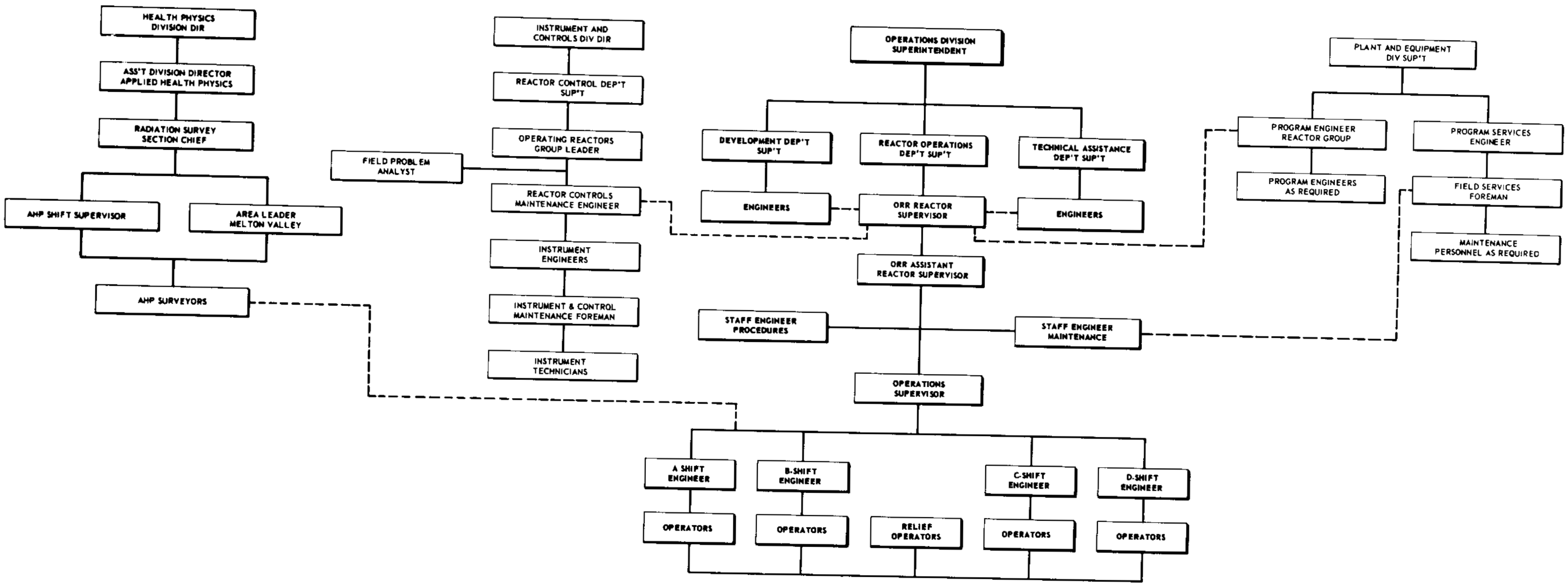

Fig. 12.3. Cooperating Divisions' Organization Chart.

required of him. He is then permitted to perform these tasks under the supervision of experienced personnel until he is judged competent to handle them independently. In addition, lectures on various pertinent topics are given and reading material assigned. The subject matter is essentially the same for both engineers and operators but is presented to the engineers at a considerably more advanced level. In the case of the operators, the initial educational work is generally begun during the three months' probationary period. It is somewhat more extended in the case of the operating engineer; he is expected to supplement his knowledge with independent study. Upon completion of training both the engineers and operators satisfy the requirements of AEC Manual Chap. 8401.

By the end of three months the operator should be qualified to perform routine duties. Training after the first three months is less intensive but continues until the operator is able to carry out his tasks independently in a competent manner.

Training of the operating engineer takes somewhat longer; generally, at least six months is required before he is given independent responsibility. This is because, in addition to being thoroughly familiar with the mechanical characteristics of the reactor, its control circuitry, and its behavior under all conditions, he must be familiar with the operator's job also.

As an aid to the trainees, a manual has been prepared which contains pertinent questions concerning the reactors. Also included is a list of supplemental reference material which aids in obtaining and understanding the answers. The operating engineer is expected to be thoroughly familiar with this material at the completion of his training period, and the operator is expected to be familiar with certain parts of it.

The senior supervisory personnel usually are drawn from the ranks of the shift engineers. The normal route is for a promising shift engineer to graduate to day work, where he is given increasing administrative responsibility and on-the-job training.

Of the current operating staff for the ORR, the reactor supervisor has 15 years of reactor operating experience; his assistant, an ORSORT graduate, has 8 years. One of the day-shift engineers has 21 years of experience and the other at least 3 years; and the shift engineers range from 1 to 15 years' experience. Of the 18 operators currently employed, only 2 have less than 5 years' experience.

12.4 Technical-Support Organizations

Two staff organizations are available in the Division to provide technical support for the operating department: the Development Department and the Technical Assistance Department.

The Development Department consists of a group of seven reactor engineers who are responsible for long-range improvements in operating techniques and safety. The personnel of this group, who are responsible to the Division superintendent, are all graduate engineers with either long experience in the reactor engineering field or ORSORT training, or both. The department is available for consultation and assistance at all times and handles such matters as heat transfer, criticality, and shielding calculations. It also assists in the development and assessment of operating procedures and is responsible for many technical details concerning operation.

The Technical Assistance Department also consists of a group of seven reactor engineers having qualifications similar to those in the Development Department. The primary function of this group is the evaluation of in-reactor experiments from the standpoint of personnel safety and operational compatibility. The Technical Assistance Department, like the Development Department, is always available for advice and consultation on safety and operation, but, in general, restricts itself to day-to-day routine matters.

In addition to these staff organizations, the Division has available upon request the services of the entire Oak Ridge National Laboratory for assistance and consultation when specific problems arise. Among the divisions most often called upon for advice with respect to reactor engineering, experiment design, reactor safety, and other matters pertaining to operations are the following, with which close contact is maintained. Included in the listing is a brief résumé of their primary functions.

The *Plant and Equipment Division* maintains a staff of engineers with shop and maintenance personnel well qualified to analyze, design, develop, fabricate, and maintain mechanical components for nuclear research reactors and associated facilities.

Engineers have designed dies for forming, blanking, cutting, and shearing of High-Flux Isotope Reactor fuel plates; designed fuel elements for the LITR, BSR, and TSR I and II; and performed heat-transfer analyses, stress analyses, and system analyses for experimental in-pile loops for the ORR.

The Plant and Equipment Fabrication Department's technical and craft personnel represent hundreds of man-years of experience in the development of fabrication techniques and the actual fabrication of fuel elements required by ORNL's operations.

The Plant and Equipment maintenance engineers and craft personnel have repeatedly demonstrated capabilities during reactor shutdowns to remove and replace experimental equipment and faulty operating components.

The *Reactor Controls Department* of the *Instrumentation and Controls Division* was established to support the Laboratory's reactor development and reactor operations programs. The department, consisting of approximately 50 physicists, engineers, and technicians, includes groups attached to the various reactor projects and engaged in the design of reactor control and safety systems, a development group employing advanced techniques in devising improved control systems and components, an analytical group which maintains an analog computing facility for the solution of reactor dynamic and kinetic problems, and an operations group which is responsible for the maintenance and related activities in the Laboratory's research and experimental reactors.

The *Neutron Physics Division* is responsible for the operation of the Tower Shielding Reactor, the Health Physics Research Reactor, and the Critical Experiments Facility. The Division performs studies supporting the design and modification of reactors and is, therefore, qualified to analyze any irregular reactor behavior. Studies are performed to assist in the design of shields for reactors, both stationary and mobile, for high-energy accelerators, and for space vehicles. The Division also performs "in-the-field" and leakage-radiation measurements for operating reactors, maintains a Radiation Shielding Information Center for reactor, weapons, and high-energy radiations, conducts a basic reactor physics program which yields neutron cross sections needed for reactor and shielding calculations, studies neutron diffusion in moderating materials, and conducts a program to measure neutron and gamma-ray spectra.

The *Solid State Division* concerns itself in large measure with a study of radiation damage in all types of solids and has developed considerable skill in the design, construction, and operation of "in-reactor" facilities. These skills have been augmented by practical knowledge of handling radioactive material, including the measurement of many physical properties of irradiated materials. Many radiation-damage experiments and studies on defects often require radiation sources other than reactors, and the Division possesses several ^{60}Co gamma sources and ^{137}Cs sources. Electronic and heavy-particle accelerators have often been employed in the experimental work of the Division. The Division has specialists in many areas of solid state in addition to the above – for example, magnetism, low temperature, superconductivity, crystal growth, x-ray and electron microscopy, transistors, surfaces, elastic constants, mechanical properties, neutron diffraction, thermal and electrical conductivity, brittle fracture, creep, internal friction, reactor physics, and solid-state theory in general.

The *Inspection Engineering Department* is responsible to ORNL management for the pressure-containing adequacy of components in reactor, radiochemical, and nonnuclear systems. In fulfilling this responsibility it: conducts engineering review of drawings and specifications to determine compliance with specifications and established codes or code equivalents and for additional ORNL safety responsibilities where existing standards are inadequate or insufficient; establishes inspection requirements and schedules to ensure adequate periodic inspection of operating equipment and arranges, with using groups, periodic retesting as required; provides and trains the personnel to inspect such equipment to ensure compliance with code, code equivalent, and extracode requirements; witnesses or performs such inspections, initially and periodically, under professional engineering supervision; reports any failures to comply with established procedures to management; prepares and maintains inspection records for pressure-containing equipment at ORNL; recommends repairs and reinspects after repair; estimates and recommends retirement dates for deteriorated equipment; and provides periodic inspections and in situ tests of high-efficiency filter systems, including the development of filter test methods and engineering consultation in connection with their application. Members of the department's professional engineering staff are active participants in national nuclear standards-writing activities and are aware of the requirements, adequacy, and shortcomings of the applicable codes.

The *Reactor Chemistry Division* is responsible for chemical support to all ORNL reactor development programs and conducts a large research program devoted to chemical aspects of nuclear safety, including the consequences of deliberate and accidental melting of reactor fuel elements, the innocuous simulation of reactor accidents, the development of techniques for removing particulate material and iodine from gas streams, and on-site testing of filters for decontamination of reactor effluent gases. Included in the Division personnel are most of the chemists formerly engaged in chemical studies of the aqueous homogeneous reactor program; hence the Division can furnish background information on the behavior of high-temperature aqueous systems. A large staff of experts on the corrosion of aluminum and other reactor construction materials, both in the presence and in the absence of mixed pile radiations and fissioning uranium, is currently engaged in studies of the chemistry of water-cooled reactors and the development of techniques for removal of radioactive corrosion products and fission products from primary reactor coolant streams. Expert personnel and facilities are also available for the identification of particulate materials or other solids found in the course of reactor operation.

The *Health Physics Division* is responsible for personnel monitoring and for maintaining personnel exposure records; is responsible for radiation monitoring of all operations, both routine and experimental; is responsible for radiation monitoring in the environs; provides round-the-clock surveillance of radiation exposure conditions and of environmental contamination; reviews sections of Standard Operating Procedures relating to hazards control and advises all divisions on the safe conduct of their operations; participates in the radiation exposure evaluation aspects of hazard studies for site selection, reactor and experiment design, waste management, and emergency planning; cooperates in radiation safety training programs in all divisions and conducts health physics training for outside groups as required; and participates in graduate education of health physics specialists at Vanderbilt University and the University of Tennessee. Research conducted by the Division embraces many direct and indirect problems of radiation measurement, dose measurement and estimation, radiation effects, and environmental effects.

The *Chemical Technology Division* is engaged in the development of chemical separations processes for use in the Commission's program. This includes the safety aspects of radiochemical plant design and operation. The Division has primary responsibility, with the Metals and Ceramics Division, for the Transuranium Processing Facility, which will prepare and process the HFIR irradiation targets; is engaged in the development of waste disposal technology for highly radioactive liquid waste effluents; builds and operates radiochemical plants; and operates hot cells for chemical separations development work.

The *Metals and Ceramics Division* carries out a broad program of research and development on metallic and ceramic materials, with emphasis on those materials used in nuclear reactors. The Division is well equipped and experienced in providing support to a wide variety of reactor projects in the areas of materials evaluation and selection, component fabrication, testing before and after service, and failure analysis. A continuing program is maintained for the development of advanced reactor fuel and control elements. The Division has had a major role in fuel development for many reactors, particularly those requiring significant advancement in the technology of aluminum-base fuels, from the pioneering MTR through the HFIR. Many types of fuel and control elements can be fabricated on both developmental and limited-production bases. Capabilities include traditional and advanced techniques for melting and casting, powder metallurgy, ceramic fabrication, extrusion, rolling, and welding and brazing. For the examination and evaluation of materials and components, including those that are highly radioactive and require remote techniques, the Division can apply new and established techniques in metallography, mechanical properties, corrosion, x-ray diffraction, nondestructive testing, and other specialties. Constant cognizance and, in many cases, liaison are maintained with the major reactor projects of the free world.

The *Analytical Chemistry Division* is responsible for performing the chemical analyses required in the various programs of the Laboratory. A large part of such requirements originates in various phases of reactor programs, particularly in metallurgical and chemical studies for reactors, in chemical processing studies, and in isotope research and production. In support of this function, the Division is responsible for developing the analytical methods required to perform these analyses. In addition, the Division is responsible for carrying out research programs in the field of analytical chemistry as designated by the Atomic Energy Commission.

The *Reactor Division* is a group of over 300 scientists, engineers, technicians, and design draftsmen with special skills and experience in the design, evaluation, development, and operation of several types of advanced nuclear reactors, including graphite-moderated, natural-uranium; pressurized-water; aqueous-homogeneous; molten-salt; and gas-cooled uranium oxide. In the course of reactor development at ORNL, the Reactor Division has developed facilities and has accumulated equipment for handling a wide range of reactor engineering problems. Personnel of the Reactor Division were responsible in whole or in part for the mechanical and nuclear design of all operating ORNL reactors.

The Division is organized into several departments, each specializing in a special phase of reactor technology. Mechanical design of reactor systems is the specialty of the Design Department, which is made up of ~36 engineers and 30 draftsmen. The Analysis Department is composed of 22 professional people, including 7 Ph.D's, and is responsible for the neutronic design, analysis and evaluation, and shielding calculations for all the reactors undertaken by the Division. Personnel in two Engineering Development Departments, with a combined total of 39 engineers and 18 technicians, are competent and experienced in the development of reactor systems as well as reactor components. Special groups in the Development Departments are especially qualified for the development of aqueous liquid-metal and molten-salt pumps, rotating compressors for gases, and reactor core hydrodynamics. The Engineering Science Department specializes in two areas: one in basic heat transfer and fluid flow and the other in stress analysis, both theoretical and experimental. This department is made up of a total of 16 engineers and 12 technicians. The Operations Department and the Irradiation Engineering Department, with a combined total of 21 engineers and 19 technicians, have operated, managed maintenance, and analyzed the performance of several experimental reactors, as well as a large number of experiments and loops in other operating reactors. Finally, the Special Projects Department, with a total of 26 engineers and 3 technicians, has high-level capabilities in several specialized engineering areas, including space reactors, Army and Maritime reactors, and reactor safety analysis.

12.5 Method of Operation

As is the case with other reactors operated by the Operations Division, the ORR is operated through the use of carefully prepared, written standard procedures. These procedures are designed to ensure that the operation of the reactor is carried on in a safe well-regulated manner. The operating procedures describe in detail the steps required for all routine operations and for as many nonroutine operations as can be anticipated. In addition to the step-by-step detail, the procedures supply information concerning the need for the particular method of operation, special hazards which may be encountered, and references to various types of descriptive material such as blueprints or component operating manuals.

The operating procedures are written by Operations Division personnel and are carefully reviewed by senior staff members of the Division. All procedures are numbered and maintained in books or procedure manuals for ready reference by the operating personnel. As procedure revisions become desirable, or as the necessity for new procedures arises, these are prepared and, after review and acceptance by appropriately designated specialists and by the Superintendent of the Reactor Operations Department, are then made part of the procedure manual.

In some cases, where the operation is quite complex or where errors cannot be tolerated, the procedure is supplemented by a checklist. Most of these checklists are to be completed by the operator and reviewed by the shift engineer. In some instances, however, the shift engineer himself is required to complete the checklist. A few examples of the operations requiring checklists are reactor startup, reactor shutdown, and daily shift checks of equipment performance and servicing. There are also checklists for certain major maintenance operations.

At times, temporary procedures are required when nonroutine operations or experiments are performed with the reactor. Such procedures are prepared in advance and approved, as in the case of new or revised procedures. During shutdowns many operations may be performed, and in such cases a temporary or shutdown procedure is written in advance to ensure that no work is forgotten and that all standard procedures are followed before the reactor is again started up.

Emergency procedures are provided for those types of malfunctions which can be anticipated. These include methods of coping with contamination or radiation incidents and fires. In addition, procedures for handling such emergencies as loss of electrical power, loss of ventilation, and instrument malfunction, among others, have been prepared. Closely associated with this is the Laboratory-wide emergency plan, which details the action to be taken in case of a serious emergency.

Communication from shift to shift is accomplished by means of the ORR log book, in which the details of the work of the shift are recorded, and which is therefore a minute history of the operation. In general, the information contained in the log book can be summarized under the following headings:

1. Operations	5. Maintenance
2. Shutdowns	6. Service to Research
3. Troubles	7. Sample Irradiations
4. Routine Checks	8. Miscellaneous

The strip charts from the various reactor instruments serve to supplement this information.

In cases where it is practical, procedures are written to describe the various maintenance operations. This is particularly true in the case of the routine maintenance of the instrumentation and controls complement and with respect to routine lubrication and maintenance of mechanical equipment. In critical cases, instrument test procedures are supplemented by the use of checklists and may be considered a part of the operating procedures. An IBM card system is used to keep abreast of routine mechanical maintenance and the stocking of spare parts.

12.6 Internal Safety Reviews

Aside from the interdepartmental safety reviews implied in the method of operation described above, the Laboratory maintains a number of standing review committees which report to the Laboratory Director and whose functions are to provide internal safety surveillance independent of the various operating and research divisions.[2] These committees are composed of senior members of the ORNL staff selected for their competence in the particular field, but in general not directly associated with the projects they review. Of these committees, three are concerned with the ORR operation. These are as follows:

The *Reactor Operations Review Committee* reviews summaries submitted by the supervisors of all ORNL reactors of their yearly operations, including such operational data as power levels, shutdown experience, and in particular an analysis of unusual occurrences. Consideration is given by the committee to the condition of the operating procedures, maintenance program, personnel changes, and reactor mechanical details which could affect the reactor shutdown margin.

In connection with these reviews, the committee conducts inspections of the reactor. During these inspections, a special point is made to observe reactor startup and shutdown procedures and to scrutinize the log book and other procedural material. At the time of the formal review, the committee may question the operating group concerning any of the items observed during the inspection, contained in the report, or otherwise brought to their attention. As a result of this review, specific recommendations are made to the Laboratory management by the committee concerning continued operation of the reactor.

In addition to this review function, concurrence of the Reactor Operations Review Committee is required before any changes are made in reactor operation which may have a significant adverse effect on safety.

The *Reactor Experiment Review Committee* reviews (from the standpoint of personnel and equipment safety and that of ensuring continuity of operations) any new or unusual experiments proposed for insertion into the reactor.

[2]Francois Kertesz, *The Auditing of Reactor Safety at the Oak Ridge National Laboratory*, ORNL-TM-612 (July 8, 1963).

Experiments proposed for the reactor are first carefully examined for safety by the Technical Assistance Department of the Operations Division.[3] It is attempted at this level to resolve any problems regarding safety and to produce a design which meets the necessary requirements. Once agreement has been reached, the experiment may be approved by the Technical Assistance Department for insertion into the reactor. If any significant hazard existed, even though it has been corrected by design, the experiment is submitted with appropriate recommendations to the Experiment Review Committee for further review. When the committee concurs that the experiment is safe, the experiment may be inserted into the reactor. The committee may make recommendations and conditions on design and operation of the experiment.

In addition to examining new experiments, the committee periodically reviews all the experiments in the reactor to ensure that they are being handled according to its recommendations. The committee also has the prerogative of overriding the approval of the Technical Assistance Department and requiring additional review of any experiment if it deems this necessary.

[3]C. D. Cagle, *General Standards Guide for Experiments in ORNL Research Reactors*, ORNL-TM-281 (Aug. 20, 1962).

The *Criticality Review Committee* has jurisdiction over operations which involve the handling, storage, and transportation of significant quantities of fissionable material. Reactor fuel within a reactor core is specifically exempted from this; however, procedures for handling fuel before insertion and after removal must be approved by this committee. The committee acts in many respects as a consulting group and gives assistance in problems involving criticality. It also conducts an annual review of each facility to ensure that approved procedures are being followed.

Finally, the Laboratory has established, as a staff function of the Director's office, a *Safety and Radiation Control Department*. This organization establishes, on behalf of Laboratory management, policy with respect to radiation protection and ascertains that this policy is met at all times. It promulgates criteria, for example, for facility containment, and serves a liaison function between the various Laboratory divisions. Staff members of the Safety and Radiation Control Department are assigned responsibilities for closely following the activities of those Laboratory divisions which handle significant quantities of radioactive materials. Specialists in key elements of the radiation safety program, such as containment, waste disposal, criticality, and reactor safety, are on the staff of the Director of Safety and Radiation Control.

Appendix A

GENERAL CHARACTERISTICS OF THE ORR

Component Design Data

Reactor vessel	
Height, ft	17.7 (includes coolant exit lines manifold, which is buried in concrete)
Diameter, ft	5.3
Wall thickness, in.	Varies from 0.75 to 1.75
Materials	Aluminum (A54S and A52S) and stainless steel (304)
Design working pressure, psig	28
Test pressure, psig	
Upper tank section	60
Lower tank section	38
Operating pressure, psig	
Upper tank section	38
Lower tank section	13
Pools	
Width, ft	10
Length, ft	
Reactor pool	21
Center and west pool	39
Depth, ft	
Center and west pool	$26\frac{1}{2}$
Reactor pool	$28\frac{2}{3}$
Distance from pool surface to pressure vessel top	$13\frac{1}{2}$
Distance from pool surface to reactor core center	24
Pool liner	
Material	Aluminum plate
Thickness, in.	$\frac{1}{4}$
Water volume, gal	
Reactor pool	$\sim$45,000
Center and west pool	$\sim$80,000

Core	
Configuration	Variable within a rectangular parallelepiped
Core lattice	7 × 9
Overall dimensions, in.	27 long × 22 wide × 24 high
Typical core dimensions, in.	21 long × 12.7 wide × 24 high, with remaining space occupied by beryllium reflector elements and experiments
Fuel load, kg of ^{235}U	4.5 to 5.6; varies with number and type of experiments
Reflector	
Material	Beryllium metal, backed by H_2O
Thickness of beryllium, in.	3 to 6
Coolant	H_2O
Shim-safety rods	6
Overall composition, vol %	
Water	60.4
Aluminum	39.4
Uranium	0.2
Fuel elements	
Type	MTR-type element with curved fuel plates and square end boxes
Number	Varies with reactor loading; 24 elements and 6 shim-safety rods are typical
Weight of ^{235}U per element, g	240 (initially)
^{235}U enrichment, %	~93
Fuel plates per element	19
Element dimensions, in.	
Length	$38\frac{3}{8}$
Width (through side plates)	2.996
Width (through outside fuel plates)	3.068
Coolant gap	0.104 (minimum)
Inside fuel plates, in.	
Thickness (overall)	0.050
Length (overall)	$24\frac{5}{8}$
Clad thickness	0.015
Core (alloy) thickness	0.020
Core (alloy) length	$23\frac{5}{8}$
Width (before bending)	2.8

Fuel elements (continued)	
Outside fuel plates, in.	
Thickness (overall)	0.065
Length (overall)	$27\frac{1}{8}$
Clad thickness	0.0225
Core (alloy) thickness	0.020
Core (alloy) length	$23\frac{5}{8}$
Width (before bending)	2.8
Plate radius of curvature, in.	5.5
End box locating pads, in.	
Distance (through flat sides)	3.032 to 3.034
Distance (concave to convex side)	3.186 to 3.188
Total weight of fuel element, kg	4.42
Fuel composition	Uranium-aluminum alloy, 21.7 wt % U
Cladding	Aluminum (ASTM B209-65, alloy 1100)
Plate temperature, °F	
Surface	210 (design maximum)
Core	224 (design maximum)
Shim rods (No. 3 through No. 6)[1]	
Configuration	Modified rectangular tube used for combination safety, shim, and regulating functions
Number	12 maximum; 6 typical
Weight of ^{235}U per shim rod, g	141 or 154 (initially)
^{235}U enrichment, %	~ 93
Fuel plates per shim rod	14
Dimensions for shim-rod assembly, in.	
Length	$117\frac{11}{16}$
Width (through side plates)	2.838
Width (through heel and toe plates)	3.029
Center line of cadmium to center line of fuel	$\sim 29\frac{1}{4}$
Bottom of cadmium to top of fuel	~ 2

[1]Number 1 and No. 2 shim rods may have aluminum section instead of fuel.

Shim rods (No. 3 through No. 6) (continued)

Upper section, jacketed cadmium insert	
Length (overall)	$30\frac{3}{4}$
Length (cadmium)	$30\frac{1}{2}$
Outside dimensions (width)	2.345 (square)
Insert thickness	0.080
Cadmium thickness	0.040
Jacket thickness	0.020
Radius of curvature of convex and concave sides	5.5
Follower section, fuel follower	
Fuel plates	
Thickness (overall)	0.060
Length (overall)	$24\frac{5}{8}$
Clad thickness	0.020
Core (alloy) thickness	0.020
Core (alloy) length	$23\frac{5}{8}$
Coolant gap	0.117
Width (before bending)	2.8
Radius of curvature	5.5
Nominal rod worth ($\Delta k/k$)	0.30 estimated (4 rods)
Withdrawal rate, in./min	5 (maximum)
Withdrawal distance, in.	$29\frac{1}{4}$
Magnet release time, msec	<5
Push-rod response time, msec	<25
Time of flight, msec	<320
Magnet current (steady), ma	~ 76
Drop current, ma	~ 37

Shim-rod drives

Type	External lead screw and nut for each rod
Motor	$\frac{1}{8}$-hp 1800-rpm, 120-v, single-phase, 60-cps reversible induction motor (condenser run)
Position indicator	Two-speed selsyn ($\sim$0.01-in. accuracy)
Traverse, in.	30

Fission chamber drives

Motor	16-w, 1800 rpm, ac low inertia servo, 115-v, 60-cycle
Traverse, in.	72
Speed, fpm	8

Reactor primary cooling system	
Coolant	H_2O
Total volume, gal	70,000
Flow rate, gpm	$\sim$18,500
Decay-tank capacity, gal	10,000
Main pumps	
Number	Three
Rating	250 hp, 6000 gpm
Shutdown pump capacity, gpm	1000
Emergency pump capacity, gpm	1000
Shell and tube heat exchangers	
Number	Four
Volume per exchanger, gal	2500
Shell and tube heat transfer capacity, Mw	29.2 (design maximum at 30 Mw)
Total primary system pressure drop, psig	54
Exit pressure from main pumps, psig	65
Inlet pressure to main pumps, psig	11
Inlet pressure to reactor tank, psig	38
Exit pressure from reactor tank, psig	13
Reactor pressure drop (ΔP), psi	$\sim$25
Coolant temperature, °F	
Reactor-tank inlet	117 ± 0.25 (design maximum 120)
Reactor-tank exit	128 (design maximum 131.5)
ΔT	11.0
pH	5.5 to 6.5
Resistivity, megohms	$\sim$1.0
Activity, dis/min	$\sim$50,000
Bypass filters	
Number	Two
Capacity, gpm	100
Degasifier capacity, gpm	50
Experiment-facility coolant pumps	
Number	Two
Capacity, gpm	400
Experiment-facility coolant flow, gpm	330
Demineralizers	
Number	Two
Capacity, gpm	80
Cation beds (Amberlite IR-120)	
Number	Two
Volume, ft^3	30

Reactor primary cooling system (continued)	
Anion beds (Amberlite IRA-401)	
Number	Two
Volume, ft^3	36
Mixed beds	
Amberlite IR-120	
Number	Two
Volume, ft^3	12
Amberlite IR-401	
Number	Two
Volume, ft^3	18
Minimum coolant flow required to remove decay heat, gpm	500
Water flow with three pony motors, gpm	2000 to 3000
Reactor secondary cooling system	
Coolant	H_2O
Total volume, gal	70,000
Flow rate, gpm	10,500
Pumps	
Number	Three
Rating	300 hp, 3500 gpm
Cooling tower basin capacity, gal	60,000
Circulating time, sec	386
Cooling tower	
Heat transfer capacity, Mw	30.78
Design inlet H_2O temperature, °F	104
Design outlet H_2O temperature, °F	78
Design flow rate, gpm	10,500
pH	6.25 to 7.5
Resistivity	Not measured
Activity	None
Pool primary cooling system	
Pool system capacity, gal	160,000
Flow rate, gpm	Varies (620 to $\sim$750 typical)
Pumps	25 hp, 900 gpm
Heat exchanger capacity, Mw	1.5
Pool-water inlet minimum temperature, °F	82
Pool-water outlet maximum temperature, °F	95
ΔT (normal), °F	7
Demineralizer flow rate, gpm	75

Pool primary cooling system (continued)

Demineralizer pump	$7\frac{1}{2}$ hp. 100 gpm
Cation bed volume, ft^3	40
Anion bed volume, ft^3	35
Storage tanks	
Number	Two
Capacity, gal	30,833
Fill and drain pump rating, gpm	300
pH	5.8 to ~6.2
Resistivity	0.75×10^6 ohms
Activity, dis/min	~1000
Filter flow, gpm	150
Reservoir tank, gal	400
Pump from reservoir tank, gpm	100
Gamma absorption heating, kw	1100 (calculated)
Fast neutron heating, kw	35 (calculated)
Natural convection from reactor tank, kw	225 (calculated)
Heat absorbed in pool, kw	~800
Normal control temperature at inlet to pool, °F	88
Neutron flux in the pool just outside the reactor tank wall, neutrons cm^{-2} sec^{-1}	1.0×10^{13}

Pool secondary cooling system

Flow rate	Varies
Pump	40 hp, 900 gpm
Total volume, gal	5270
Water volume in basin and sump, gal	1860
pH	6.2 to 6.6
Resistivity	Not measured
Activity	None

Building

Length, ft	108
Width, ft	100
Crane width, ft	60
Height, ft	
Main floor to wing	36
Main floor to crane bay	71
Basement area, ft^2	11,400
Main floor area, ft^2	9500
Second floor area, ft^2	6240
Third floor area, ft^2	6800

Hot cells	
Number	Two
Wall	
Material	Barytes concrete
Thickness, ft	3.5
Building service	
Plant process water	6-in- main provides water at 50 psig pressure
Demineralized water, gpm	10 continuous, 50 intermittent
Dry compressed air	160 cfm at 90 psig
"Hot" off-gas exhaust, cfm	500 (20 to 30 in. H_2O suction)
Normal electric power	800 kw, 440 v, three phase, 60 cps (does not include instrument power, main pumps, and cooling fans)
Emergency electrical power	350 kw, 440 v, three phase, 60 cps
Shielding	
Designed maximum radiation at normal personnel access points, mr/hr	0.75
Calculated direct radiation through the water (24 ft), mr/hr	7.5
Calculated ^{16}N at pool surface, mr/hr	100 (maximum), normally less than 7
Radiation in pipe chase, r/hr	20
Nuclear design data	
Steady-state power, Mw	30
Thermal neutron flux (average), neutrons cm^{-2} sec^{-1}	1.4×10^{14}
Thermal neutron flux (maximum), neutrons cm^{-2} sec^{-1}	3.4×10^{14}
Epithermal neutron flux (average), neutrons cm^{-2} sec^{-1}	3.5×10^{14}
Critical loading (× core), kg	1.28
Control requirement (typical) $(\Delta k/k)$, %	
Fuel depletion/low σ_a f.p.	33×10^{-3}
Xenon	39×10^{-3}
Samarium	10×10^{-3}
Temperature	4×10^{-3} (70 to ~125°F)
Experiment	50×10^{-3}
Margin for operation	10×10^{-3}
Total control requirement $(\Delta k/k)$, %	146×10^{-3}
Total control available $(\Delta k/k)$, %	300×10^{-3}
Prompt neutron lifetime, sec	$\sim 10^{-4}$
Gamma heating, w/g	3 to ~10

Nuclear data	
Temperature coefficient, % $\Delta k/k$ per °F	7.50×10^{-3}
Void coefficient (4 × 7 core), % $\Delta k/k$	
HB-1 drained	−0.035
HB-1 and HB-2 drained	−0.049
HB-1, HB-2, and HB-3 drained	−0.074
HB-2 and HB-3 drained	−0.054
HB-3 drained	−0.047
HB-1, HB-2, HB-3, HB-5, and HB-6 drained	−0.127
HB-2 drained	−0.017
HN drained	−0.470
Thermal neutron flux, neutrons cm^{-2} sec^{-1}	
Highest neutron flux	5×10^{14}
Average neutron flux	1.6×10^{14}
Engineering facilities HN-1 and HN-2	7×10^{13}
F-8	$1 \sim 2 \times 10^{14}$
HN-3	6.5×10^{13}
B-1 (gas-cooled loop 1)	2.5×10^{13}
Poolside facility	3×10^{13}
A-2 (experiment facility)	7×10^{13}
F-2	1.2×10^{14}
Fast neutron flux , neutrons cm^{-2} sec^{-1}	
C-3	5×10^{14}
C-7	3×10^{14}
B-8	9×10^{13}
Beam hole flux, neutrons cm^{-2} sec^{-1}	
1-1B-1, thermal	$\sim 10^{9}$
HB-2, fast	$\sim 1 \sim 2 \times 10^{8}$
HB-3 and HB-4, thermal	10^{9}
HB-5, thermal	10^{9}
HB-6, thermal and fast	10^{9}
Poolside facility	10^{13}
Heat transfer data	
Coolant volume in core, gal	~18
Flow rate, gpm	18,500
ΔP across core, psi	24.5
ΔT across core, °F	11.0
Coolant speed between fuel plates, fps	32
Maximum heat transfer rate, Btu ft^{-2} hr^{-1}	6×10^{5}

Radiation level (11-5-66), mr/hr	
Reactor demineralizer	
Anion unit	460
Cation unit	200
Mixed bed	20
Pool demineralizer	
Anion unit	26
Cation unit	10
Pump house	
No. 1 pump	250
No. 2 pump	270
No. 3 pump	300
Heat exchanger	
Entrance	10
North side of pit	10
South side of pit	10
Pool surface	5

Appendix B

SPECIFICATIONS

Specification for ORR Fuel Elements

Specification No. 4, August 1966

1.0 Scope

1.1 This specification covers the fabrication, inspection, and packaging of the aluminum-clad plate-type fuel elements for the ORR.

2.0 Drawings[1]

2.1 The following drawings are considered a part of this specification;

2.1.1 M-20394-EJ-001-D 19-Plate Fuel Element – Assembly and Finish Machining

2.1.2 M-20394-EJ-002-D 19-Plate Fuel Element – Detail Sheet No. 1

2.1.3 M-20394-EJ-003-D 19-Plate Fuel Element – Detail Sheet No. 2

2.1.4 M-20394-EJ-004-D 19-Plate Fuel Element – End Box Casting

3.0 Description of Elements

3.1 Each fuel element is to be fabricated according to the drawings listed in Sect. 2.0. The fuel assemblies are to be mechanically joined. Each assembly is to consist of 19 composite plates containing ^{235}U fuel in the form of an aluminum-clad uranium-aluminum alloy. The cladding shall be metallurgically bonded to the fuel core and the fuel core hermetically sealed.

4.0 Materials

4.1 In addition to any other requirements, all materials specified below shall be certified by the Seller to contain less than 10 ppm boron, 80 ppm cadmium, and 80 ppm lithium. The quantities of these impurities (boron, cadmium, and lithium) in the finished element shall not exceed these values.

4.2 Fuel Core Alloy

4.2.1 Uranium

The uranium melting stock for the alloy shall contain a minimum of 93 wt % ^{235}U and shall contain less than 300 ppm of carbon.

4.2.2 Aluminum

The aluminum melting stock for the fuel alloy shall conform to ASTM specification B209-65, alloy 1100, or a higher purity grade.

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[1]All drawings specified in this document are ORNL drawings. The Seller will furnish a minimum of two copies of the equivalent drawings and a set of reproducible drawings applicable to his mechanical assembly procedure. All finished dimensions and tolerances shall be identical to those shown in the drawings above.

4.3 Core Frames and Cladding

The aluminum for the core frames and cladding shall conform to ASTM specification B209-65, alloy 1100. The core frames shall be one piece; frames made of multiple pieces welded together are not acceptable.

4.4 Side Plates

The aluminum for the side plates shall conform to ASTM specification B209-65, alloy 6061-T6.

4.5 End Adapters

The end adapters shall be made of aluminum conforming to ASTM specification B26-65, alloy SG70A-T6, for sand castings or ASTM specification B108-65, alloy SG70A-T6, for permanent mold castings.

4.6 Combs and Pins

The combs and pins shall be made of aluminum conforming to ASTM specification B209-65, alloy 6061 or alloy 1100.

4.7 Aluminum Welding Rods

Aluminum welding rods shall conform to ASTM B285-61T type ER4043.

5.0 Fuel Loading

5.1 Each fuel element shall contain 240 g $\pm 2\%$ of ^{235}U. Each fuel plate shall contain 12.63 g $\pm 4\%$ of ^{235}U. The value of ^{235}U in the element is to be determined by a uniform, statistically sound sampling procedure proposed by the Seller and approved by the Company. The weight of each core shall be measured and recorded to within 0.01 g along with the calculated value of the ^{235}U content. Fuel content may be determined by the Company by the use of reactivity measurements.

6.0 Welding

6.1 General

The upper and lower adapter castings shall be welded to the fuel elements by an inert-gas-shielded arc-welding process. A procedure shall be qualified and submitted, together with qualification test specimens and workmanship standards (see below), to the Company for approval prior to welding onto the fuel elements. Evidence of qualification of welders shall also be submitted.

6.2 Qualification

Each procedure and welder qualification test shall include the welding of at least two test welds of each type involved. Test welds shall be inspected to the same standards as required of the production welds. One test weld shall be further evaluated to the standards for fillet welds in QN-9 (*b*) and (*c*) of Sect. IX of the 1965 ASME Boiler and Pressure Vessel Code, except that the joint designs, materials, and dimensional limits of the test welds shall be the same as the drawing requirements for the production welds. The second procedure qualification test weld shall be sectioned, inspected, and, without fracture testing, retained as a Workmanship Standard after approval by the Company.

6.3 Inspection

Welds shall be visually examined for conformance to the Workmanship Standard and for evidence of cracks, pinholes, surface porosity, undercut, inclusions, excessive roughness, failure to blend smoothly into the base metal, inadequate penetration into the base metal, presence of abrupt ridges or valleys, and failure to meet dimensional requirements of the drawings. They shall also be examined by a commercial fluid penetrant of the post-emulsification type. Penetrant indications shall be explored to determine their cause. Any defects of the types listed above, established by visual or penetrant examination, shall be removed; and repairs should be effected by welding if required to meet this specification.

7.0 Manufacturing and Quality Control Requirements

7.1 Dimensions

Each element shall be inspected for dimensional adherence to the reference drawings listed in Sect. 2.0.

7.2 Location of Fuel Core

The location of the fuel core in each fuel plate shall meet the tolerances specified on the reference drawings in Sect. 2.0 and shall be verified by fluoroscopic examination.

7.3 Quality of Fuel Core

One randomly selected finished plate from each U-Al alloy melt is to be radiographed and examined for fuel location, uranium homogeneity, and inclusions of uranium carbide. The presence of any uranium carbide inclusion that is larger than an area equivalent to 0.06 in. in diameter shall be cause for rejection of all composite fuel plates made from that U-Al melt, unless the Seller can demonstrate that each other plate from this melt meets this inclusion specification.

7.4 Cladding and Core Thickness Determination

The cladding and core thicknesses, as specified on the reference drawings listed in Sect. 2.0, are to be determined for a randomly selected fuel plate from the first group of plates processed. (A plate which has been rejected for reasons not affecting the clad and core dimensions may be used for this determination.) Also, one randomly selected plate from each group of plates processed thereafter is to be sectioned for the clad and core thickness determinations. If any plate fails to meet specifications, two more plates randomly selected from the same group are to be sectioned for thickness measurements. If any of these plates fail to meet specifications, all plates manufactured or in the process of manufacture since the last acceptable sectioning are to be rejected. The plates are to be sectioned according to the procedure outlined in the proposed ANS Standard "Quality Control for Plate Type Uranium-Aluminum Fuel Elements."

7.5 Cladding Temper

Each fuel plate shall be rolled by a combination of first, hot rolling and second, cold rolling. The final reduction of the fuel plate thickness shall be accomplished by cold rolling and shall not be less than 15% nor greater than 25%.

7.6 Metallurgical Bond

The existence of a good metallurgical bond between each component of the fuel plate shall be verified by heating each fuel plate to 500°C for a period of 1 hr prior to the final 15 to 25% cold roll reduction. Plates exhibiting visible raised or blistered areas shall be rejected.

7.7 Alignment, Twist, and Camber

The finished assembly is to be inspected to determine whether or not it fits into a rectangular parallelepiped whose dimensions are $^1/_{64}$ in. greater than the maximum height and width, perpendicular to the longest dimension, of an aligned fuel assembly as shown on drawings listed in Sect. 2.0 of this specification. Fuel assemblies which do not meet this requirement shall be rejected.

7.8 Finished Fuel Plates

7.8.1 After rolling, a serial number shall be vibratooled on each fuel plate for identification as to heat number and ingot number. The vibratooled identification shall be over the unfueled region of the plate. Fuel loading and location of each plate in the element will be shown in the final inspection reports.

7.8.2 Surface Finish

Fuel plates with scratches, pits, or marks more than 0.003 in. in depth are not acceptable. Plates with dents greater than 0.012 in. in diameter and/or more than 0.012 in. deep are not acceptable.

7.8.3 General Cleanliness

The Seller shall take all necessary precautions to maintain a high standard of cleanliness during fabrication and assembly to ensure that no foreign materials or corrosion products are present in the finished elements. All surfaces must be free from moisture, dirt, oil, organic compounds, scale, paint, graphite, or other foreign matter. Use of graphite for marking purposes is prohibited. The use of abrasives for cleaning the fuel plates or for any other purposes is prohibited, as is any procedure which removes more than 0.0001 in. of aluminum from the surface of the finished fuel plates. If any chlorine-bearing material is used for cleaning, it must be completely removed following the cleaning procedure.

7.8.4 Surface Contamination

The Seller shall verify that the surface contamination of the fuel plates is less than 5 μg of uranium per square foot. If the manufacturing process requires the drilling of plates, the drill bit shall be examined for alpha-particle contamination following such drilling. If alpha contamination is present, the plate or plates shall be reexamined by fluoroscopic or x-ray procedures to determine which fuel core or cores were penetrated by the drill. The plates with penetrated cores shall be rejected.

7.9 Mechanically Bonded Fuel Elements

The mechanical joints shall have a strength (as demonstrated by a pull test of sample elements) of not less than 200 lb/linear inch per fuel plate.

8.0 Packaging and Shipping

8.1 Packaging

Immediately following the Seller's final inspection and acceptance, the fuel elements shall be placed in polyethylene bags, which will be sealed moisture-tight and dust-tight. The polyethylene shall be at least 0.010 in. thick.

8.2 Shipping

The Seller shall provide shipping containers and shall have full responsibility for the fuel elements while in transit to the site designated by the Company.

9.0 Material, Records, and Reports

9.1 A certified copy of inspection and test records covering the items listed below shall be supplied to the Company. A duplicate copy of the records of each fuel assembly shall be included in the shipping container with the assemblies.

9.1.1 Certified inspection records of each fuel element showing the principal dimensions, the adherence to the minimum water channel thickness between plates, and data regarding the total alignment, twist, and camber of each element, as specified on the drawings listed in Sect. 2.0 of this specification.

9.1.2 The location and serial number of each plate within each element, the calculated fuel loading of uranium and ^{235}U in each fuel plate, and the total calculated loading of uranium and ^{235}U and the serial number of each fuel assembly. The fuel loadings are to be as specified in Sect. 5.0 of this specification.

9.1.3 Radiographs of the representative fuel plate from each U-Al alloy melt, as prescribed in Sect. 7.3 of this specification.

9.1.4 Data on the examination of the surfaces of the element, as specified in Sect. 7.8 of this specification.

9.1.5 Data regarding the location of fuel cores, as prescribed, including the radiographs, in Sect. 7.2 of this specification.

9.1.6 Results of the cladding and core thickness determination, as specified in Sect. 7.4 of this specification.

9.1.7 Results of the inspection of bonding in each fuel plate, as specified in Sect. 7.6 of this specification.

9.1.8 A certified report of the chemical and isotopic analysis of each heat of fuel alloy, as specified in Sect. 4.0 of this specification.

9.1.9 A certified report of the chemical analysis of all other materials used in the fabrication of the fuel element, as specified in Sect. 4.0 of this specification.

9.1.10 Results of the pull test regarding the strength of the mechanical joints, as prescribed in Sect. 7.9 of this specification.

9.1.11 A certificate of compliance for each element, stating that the element meets all requirements of the contract.

10.0 Manufacturing Procedures

10.1 A complete written description of the manufacturing procedures, including shop drawings, cleaning procedures, and inspection report forms, is to be supplied to the Company by the Seller prior to initiation of any fabrication, and any subsequent changes in the procedure are also to be supplied to the Company prior to their use. In addition, inspection procedures for determining the X, W, Y, and Z measurements are to be supplied to the Company.

Specification for ORR Shim-Rod Fuel Assembly Sections

Specification No. 3, August 1966

1.0 Scope

1.1 This specification covers the fabrication, inspection, and packaging of the ORR shim-rod fuel sections.

2.0 Drawings[1]

2.1 The following drawings are considered a part of this specification:

2.1.1 M-20394-EJ-014-D Shim-Rod Fuel Section – Assembly

2.1.2 M-20394-EJ-015-D Shim-Rod Fuel Section – Detail Sheet No. 1

2.1.3 M-20394-EJ-016-D Shim-Rod Fuel Section – Detail Sheet No. 2

[1]All drawings specified in this document are ORNL drawings. The Seller will furnish a minimum of two copies of the equivalent drawings and a set of reproducible drawings applicable to his mechanical assembly procedure. All finished dimensions and tolerances shall be identical to those shown in the drawings above.

3.0 Description of Assemblies

3.1 Each assembly is to be fabricated according to the drawings listed in Sect. 2.0. The assembly is to be mechanically joined and is to consist of 14 composite plates containing ^{235}U fuel in the form of an aluminum-clad uranium-aluminum alloy. The cladding shall be metallurgically bonded to the fuel core and the fuel core hermetically sealed.

4.0 Materials

4.1 In addition to any other requirements, all materials specified below shall be certified by the Seller to contain less than 10 ppm boron, 80 ppm cadmium, and 80 ppm lithium. The quantities of these impurities (boron, cadmium, and lithium) in the finished element shall not exceed these values.

4.2 Fuel Core Alloy

4.2.1 Uranium

The uranium melting stock for the alloy shall contain a minimum of 93 wt % ^{235}U and shall contain less than 300 ppm of carbon.

4.2.2 Aluminum

The aluminum melting stock for the fuel alloy shall conform to ASTM specification B209-65, alloy 1100, or a higher purity grade.

4.3 Core Frames and Cladding

The aluminum for the core frames and cladding shall conform to ASTM specification B209-65, alloy 1100. The core frames shall be one piece; frames made of multiple pieces welded together are not acceptable.

4.4 Heel and Toe Plates

The aluminum for the heel and toe plates shall conform to ASTM specification B209-65, alloy 1100.

4.5 Combs and Pins

The combs and pins shall be made of aluminum conforming to ASTM specification B209-65, alloy 6061 or alloy 1100.

4.6 Side Plates

The aluminum for the side plates shall conform to ASTM specification B209-65, alloy 6061T6.

4.7 Aluminum Welding Rods

Aluminum welding rods shall conform to ASTM B285-61T type ER4043.

5.0 Fuel Loading

5.1 Each assembly shall contain 154.00 g ± 2% of ^{235}U. Each fuel plate shall contain 11.00 g ± 4% of ^{235}U. The value of ^{235}U in the assembly is to be determined by a uniform, statistically sound sampling procedure proposed by the Seller and approved by the Company. The weight of each core shall be measured and recorded to within 0.01 g along with the calculated value of the ^{235}U content. Fuel content may be determined by the Company by the use of reactivity measurements.

6.0 Welding

6.1 General

Welding shall be accomplished by an inert-gas-shielded arc-welding process. A procedure shall be qualified and submitted, together with qualification test specimens and workmanship standards (see below), to the Company for approval prior to welding onto the fuel elements. Evidence of qualification of welders shall also be submitted.

6.2 Qualification

Each procedure and welder qualification test shall include the welding of at least two test welds of each type involved. Test welds shall be inspected to the same standards as required of the production welds. One test weld shall be further evaluated to the standards for fillet welds in QN-9 (*b*) and (*c*) of Sect. IX of the 1965 ASME Boiler and Pressure Vessel Code, except that the joint designs, materials, and dimensional limits of the test welds shall be the same as the drawing requirements for the production welds. The second procedure qualification test weld shall be sectioned, inspected, and, without fracture testing, retained as a Workmanship Standard after approval by the Company.

6.3 Inspection

Welds shall be visually examined for conformance to the Workmanship Standard and for evidence of cracks, pinholes, surface porosity, undercut, inclusions, excessive roughness, failure to blend smoothly into the base metal, inadequate penetration into the base metal, presence of abrupt ridges or valleys, and failure to meet dimensional requirements of the drawings. They shall also be examined by a commercial fluid penetrant of the post-emulsification type. Penetrant indications shall be explored to determine their cause. Any defects of the types listed above, established by visual or penetrant examination, shall be removed; repairs should be effected by welding if required to meet this specification.

7.0 Manufacturing and Quality Control Requirements

7.1 Dimensions

Each element shall be inspected for dimensional adherence to the reference drawings listed in Sect. 2.0

7.2 Location of Fuel Core

The location of the fuel core in each fuel plate shall meet the tolerances specified on the reference drawings in Sect. 2.0 and shall be verified by fluoroscopic examination.

7.3 Quality of Fuel Core

One randomly selected finished plate from each U-Al alloy melt is to be radiographed and examined for fuel location, uranium homogeneity, and inclusions of uranium carbide. The presence of any uranium carbide inclusion that is larger than an area equivalent to 0.06 in. in diameter shall be cause for rejection of all composite fuel plates made from that U-Al melt, unless the Seller can demonstrate that each other plate from this melt meets this inclusion specification.

7.4 Cladding and Core Thickness Determination

The cladding and core thicknesses as specified on the reference drawings listed in Sect. 2.0 are to be determined for a randomly selected fuel plate from the first group of plates processed. (A plate which has been rejected for reasons not affecting the clad and core dimensions may be used for this determination.) Also, one randomly selected plate from each group of plates processed thereafter is to be sectioned for the clad and core thickness determinations. If any plate fails to meet specifications, two more plates randomly selected from the same group are to be sectioned for thickness measurements. If any of these plates fail to meet specifications, all plates manufactured or in the process of manufacture since the last acceptable sectioning are to be rejected. The plates are to be sectioned according to the procedure outlined in the proposed ANS Standard "Quality Control for Plate Type Uranium-Aluminum Fuel Elements."

7.5 Cladding Temper

Each fuel plate shall be rolled by a combination of first, hot rolling and second, cold rolling. The final reduction of the fuel plate thickness shall be accomplished by cold rolling and shall not be less than 15% nor greater than 25%.

7.6 Metallurgical Bond

The existence of a good metallurgical bond between each component of the fuel plate shall be verified by heating each fuel plate to 500°C for a period of 1 hr prior to the final 15 to 25% cold roll reduction. Plates exhibiting visible raised or blistered areas shall be rejected.

7.7 Alignment, Twist, and Camber

The finished assembly is to be inspected to determine whether or not it fits into a rectangular parallelepiped whose dimensions are $^1/_{64}$ in. greater than the maximum height and width, perpendicular to the longest dimension, of an aligned fuel assembly as shown on drawings listed in Sect. 2.0 of this specification. Fuel assemblies which do not meet this requirement shall be rejected.

7.8 Finished Fuel Plates

7.8.1 Plate Identification

After rolling, a serial number shall be vibratooled on each fuel plate for identification as to heat number and ingot number. The vibratooled identification shall be over the unfueled region of the plate. Fuel loading and location of each plate in the element will be shown in the final inspection reports.

7.8.2 Surface Finish

Fuel plates with scratches, pits, or marks more than 0.003 in. in depth are not acceptable. Plates with dents greater than 0.012 in. in diameter and/or more than 0.012 in. deep are not acceptable.

7.8.3 General Cleanliness

The Seller shall take all necessary precautions to maintain a high standard of cleanliness during fabrication and assembly to ensure that no foreign materials or corrosion products are present in the finished elements. All surfaces must be free from moisture, dirt, oil, organic compounds, scale, paint, graphite, or other foreign matter. Use of graphite for marking purposes is prohibited. The use of abrasives for cleaning the fuel plates or for any other purposes is prohibited, as is any procedure which removes more than 0.0001 in. of aluminum from the surface of the finished fuel plates. If any chlorine-bearing material is used for cleaning, it must be completely removed following the cleaning procedure.

7.8.4 Surface Contamination

The Seller shall verify that the surface contamination of the fuel plates is less than 5 μg of uranium per square foot.

If the manufacturing process requires the drilling of plates, the drill bit shall be examined for alpha-particle contamination following such drilling. If alpha contamination is present, the plate or plates shall be reexamined by fluoroscopic or x-ray procedures to determine which fuel core or cores were penetrated by the drill. The plates with penetrated cores shall be rejected.

7.9 Mechanically Bonded Fuel Elements

The mechanical joints shall have a strength (as demonstrated by a pull test of sample elements) of not less than 200 lb/linear inch per fuel plate.

8.0 Packaging and Shipping

8.1 Packaging

Immediately following the Seller's final inspection and acceptance, the fuel elements shall be placed in polyethylene bags, which will be sealed moisture-tight and dust-tight. The polyethylene shall be at least 0.010 in. thick.

8.2 Shipping

The Seller shall provide shipping containers and shall have full responsibility for the fuel elements while in transit to the site designated by the Company.

9.0 Material, Records, and Reports

9.1 A certified copy of inspection and test records covering the items listed below shall be supplied to the Company. A duplicate copy of the records of each fuel assembly shall be included in the shipping container with the assemblies.

9.1.1 Certified inspection records of each fuel element showing the principal dimensions, the adherence to the minimum water channel thickness between plates, and data regarding the total alignment, twist, and camber of each element, as specified on the drawings listed in Sect. 2.0 of this specification.

9.1.2 The location and serial number of each plate within each element, the calculated fuel loading of uranium and ^{235}U in each fuel plate, and the total calculated loading of uranium and ^{235}U and the serial number of each fuel assembly. The fuel loadings are to be as specified in Sect. 5.0 of this specification.

9.1.3 Radiographs of the representative fuel plate from each U-Al alloy melt, as prescribed in Sect. 7.3 of this specification.

9.1.4 Data on the examination of the surfaces of the element, as specified in Sect. 7.8 of this specification.

9.1.5 Data regarding the location of fuel cores, as prescribed, including the radiographs, in Sect. 7.2 of this specification.

9.1.6 Results of the cladding and core thickness determination, as specified in Sect. 7.4 of this specification.

9.1.7 Results of the inspection of bonding in each fuel plate, as specified in Sect. 7.6 of this specification.

9.1.8 A certified report of the chemical and isotopic analysis of each heat of fuel alloy, as specified in Sect. 4.0 of this specification.

9.1.9 A certified report of the chemical analysis of all other materials used in the fabrication of the fuel element, as specified in Sect. 4.0 of this specification.

9.1.10 Results of the pull test regarding the strength of the mechanical joints, as prescribed in Sect. 7.9 of this specification.

9.1.11 A certificate of compliance for each element, stating that the element meets all requirements of the contract.

10.0 Manufacturing Procedures

10.1 A complete written description of the manufacturing procedures, including shop drawings, cleaning procedures, and inspection report forms, is to be supplied to the Company by the Seller prior to initiation of any fabrication, and any subsequent changes in the procedure are also to be supplied to the Company prior to their use.

Appendix C

EXPERIENCE IN COMPUTING RESEARCH REACTOR FUEL BURNUP CONTRASTED WITH RECOVERY RESULTS

T. P. Hamrick H. F. Stringfield

Abstract

For several years ORNL has operated reactors for research purposes which use highly enriched uranium as fuel. In accordance with AEC nuclear material management procedures, it has been necessary to calculate fuel losses by burnup for material balance purposes. This paper deals with the methods used in computing burnup, including the formulas, the source of errors, and the adjustments in the formulas required to minimize errors. In addition, the amount of fuel calculated to be contained in the several hundred fuel elements that have been sent for recovery is compared with the reported recovery results.

Introduction

Consuming a fissionable material in a nuclear reactor is a process that is well understood, and good analytical models are available for the solution of this process. However, when these models are applied to an actual reactor, difficulties are always encountered. For several years, Oak Ridge National Laboratory has operated light-water-moderated research reactors which use highly enriched uranium as fuel.

It was known from the beginning that some sort of estimate of the amount of fuel consumed must be made to satisfy the Atomic Energy Commission requirements of fissionable material accountability. Later, as technology developed and reactor power levels became higher, it became evident that not only the amount of fuel consumed was important but where the fuel was consumed. Primarily this was due to the following reason. In the low-power reactor, the element life was determined not by burnup but by corrosion, so that cores were replaced in entirety, and the residual uranium was recovered by processing the core as a unit. This method is still used in many low-power reactors, whereby the amount of fuel consumed is evenly distributed to the elements which make up the core. Upon the advent of higher-power reactors, the element life became shorter, and the primary reason for reprocessing an element was the low amount of residual fuel, instead of corrosion considerations. Then the cores were not processed in entirety, and the residual uranium was recovered by processing individual or specific groups of elements, some of which had been in many core loadings. This added consideration made it necessary to know where the residual uranium was located, in order to know which element to reinsert into the core to optimize the fuel element cycle while keeping in mind the economic implications.

Burnup Calculation

One of the first position-type burnup calculations for highly enriched uranium was initiated in 1952 to be used for the Low-Intensity Test Reactor (LITR). It consisted of the following:

1. The assumption was made that the burnup rate is 1.26 g/Mwd. Of this quantity, 1.07 g undergoes fission and 0.19 g was converted to ^{236}U. Therefore, the total burnup for a core was the power level times the number of days operated at this power level multiplied by 1.26, which resulted in the total fuel consumption.
2. This total fuel consumption was then distributed among the several fuel elements and fuel-follower shim rods in the core during that particular run. This distribution depended upon the average thermal neutron flux per fuel element. In order to make the calculation of the fraction of the total burnup that should be assigned to any element easy, the sum of the burnup factors for all core positions should be unity.

Determination of Burnup Factor I

According to the above description, the fraction of the total burnup which occurred in position *i* would be

$$\frac{\text{Burnup in } i}{\text{Burnup in core}} = \frac{\overline{\Phi}_i}{\overline{\Phi}_c} ,$$

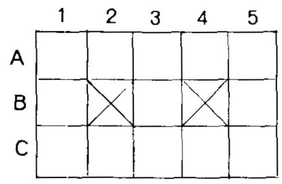
where $\overline{\Phi}_i/\overline{\Phi}_c$ was determined from a flux map. It was recognized that the above conditions held only for a short time, but they were assumed to be reasonable since, as $\overline{\Phi}_c$ changes, $\overline{\Phi}_i$ also changes proportionally. Each burnup factor was adjusted so that the sum would be unity by dividing each factor by the sum of the factors.

For our study of the evolution of burnup factors, let us assume a hypothetical core such as that shown in Table C.1. It can be seen from close analysis that the ratio of BUF-I to $\overline{\Phi}_i/\overline{\Phi}_c$ is constant for all core positions. This type of computation seemed adequate until, after much running time and many burnup calculations later, we achieved more than 100% burnup on one of our shim rods. It was realized that while the LITR had three shim rods, two were kept fully withdrawn and one was used to regulate the power (this was the rod which achieved >100% burnup). In order to correct for this abnormality, it was decided after checking back that the rod was only 66% in the core, so the following correction was made.

Determination of Burnup Factor II

The same method was used as in the determination of BUF-I, with the exception that only 66% of the burnup due the shim rod would be used. This not only lowered the calculated amount of fuel consumed in the shim rod but raised slightly the calculated amount consumed in all other core positions, since the total must remain the same.

Table C.1. Core Statistics



EBF Core

Core Position	Fuel Type	Weight of ^{235}U (g)	$\bar{\Phi}_i/\bar{\Phi}_c$	BUF-I
A-1	Fuel	200	0.833	0.05553
A-2	Fuel	180	0.952	0.06347
A-3	Fuel	160	1.071	0.07141
A-4	Fuel	180	0.892	0.05947
A-5	Fuel	200	0.773	0.05153
B-1	Fuel	180	1.178	0.07853
B-2	Shim rod	120	1.190	0.07933
B-3	Fuel	160	1.545	0.10299
B-4	Shim rod	120	1.071	0.07141
B-5	Fuel	180	1.130	0.07533
C-1	Fuel	200	0.803	0.05353
C-2	Fuel	180	0.922	0.06147
C-3	Fuel	160	1.041	0.06940
C-4	Fuel	180	0.862	0.05747
C-5	Fuel	200	0.737	0.04913
		2600	15.000	1.00000

We will now make this correction to our hypothetical EBF core, assuming that the shim rod in B-2 is fully withdrawn during operation and the power is regulated with the shim rod in B-4 (average position is 66% withdrawn). The results are given in Table C.2.

Determination of Burnup Factor III

Just prior to operation of the ORR the burnup calculation was examined, since the use of fuel during operation would be much higher, due to the higher operating power. Several problems were foreseen involving cycling of fuel. For example, in contrast with LITR practice, the xenon problem would necessitate the changeout of almost entire cores during refueling. The removed elements could be reused after xenon decay in assembling different cores. Since the weight ascribed to a partially depleted fuel element depends on the burnup calculation, the accuracy of the calculation as applied to individual elements became increasingly important. It was felt that the burnup factor depended not only upon the average thermal flux per element but also upon the quantity of fuel per fuel element. Therefore, a burnup factor for any fuel element was now determined by these two factors.

Table C.2. Calculation of BUF-II

Core Position	BUF-I	BUF-I Corrected	BUF-II
A-1	0.05553	0.05553	0.05691
A-2	0.06347	0.06347	0.06505
A-3	0.07141	0.07141	0.07319
A-4	0.05947	0.05947	0.06095
A-5	0.05153	0.05153	0.05281
B-1	0.07853	0.07853	0.08048
B-2	0.07933	0.07933	0.08130
B-3	0.10299	0.10299	0.10556
B-4	0.07141	0.04713	0.04831
B-5	0.07533	0.07533	0.07720
C-1	0.05353	0.05353	0.05486
C-2	0.06147	0.06147	0.06300
C-3	0.06940	0.06940	0.07113
C-4	0.05747	0.05747	0.05890
C-5	0.04913	0.04913	0.05035
	1.00000	0.97572	1.00000

According to this new thinking, the fraction of the total burnup which occurred in position i would be

$$\frac{\bar{\Phi}_i}{\bar{\Phi}_c} \times \frac{M_i}{M_c} = \frac{\text{burnup in } i}{\text{burnup in core}},$$

where $\bar{\Phi}_i/\bar{\Phi}_c$ is determined from a flux map as before and M_i/M_c is the ratio of fuel weight in i to fuel weight in the core. It was again recognized that the above conditions held only for a short time, but they were assumed to be reasonably accurate for a fuel cycle. The burnup factors determined in this fashion again will not add up to exactly unity. This is due to breaking up the product ratio into two ratios which are first determined independently and then multiplied together. Each burnup factor must be adjusted so that the sum will be unity by dividing each factor by the sum of the factors.

We can now look again at our hypothetical EBF core and calculate BUF-III (Table C.3).

Determination of Burnup Factor IV

The BUF-III calculation assumes that all the residual fuel in the element is exposed to the average thermal neutron flux in the element. This, in reality, is not true. Before proceeding to the determination of BUF-IV, we will briefly examine the components of the burnup factor calculation, that is, fuel distribution and thermal flux distribution.

Table C.3. Calculation of BUF-III

Core Position	Weight of ^{235}U (g)	Φ_i/Φ_c	M_i/M_c	$\frac{\bar{\Phi}_i}{\bar{\Phi}_c} \times \frac{M_i}{M_c}$	BUF-III
A-1	200	0.833	0.07692	0.06407	0.06624
A-2	180	0.952	0.06923	0.06591	0.06814
A-3	160	1.071	0.06154	0.06591	0.06814
A-4	180	0.892	0.06923	0.06175	0.06384
A-5	200	0.773	0.07692	0.05956	0.06159
B-1	180	1.178	0.06923	0.08155	0.08431
B-2 (shim)	120	1.190	0.03941[a]	0.04690	0.04850
B-3	160	1.545	0.06154	0.09508	0.09830
B-4 (shim)	120	1.071	0.03941[a]	0.04221	0.04365
B-5	180	1.130	0.06923	0.07823	0.08088
C-1	200	0.803	0.07692	0.06177	0.06386
C-2	180	0.922	0.06923	0.06383	0.06599
C-3	160	1.041	0.06154	0.06406	0.06623
C-4	180	0.862	0.06923	0.05958	0.06171
C-5	200	0.737	0.07692	0.05669	0.05862
	2600	1.000		0.96720	1.00000

[a]One other adjustment must be made in conjunction with the above calculation. If you will notice, the shim rods are corrected by the factor 0.854. This was an experimentally determined correction factor to correct for the fuel in the shim-rod follower not being fully inserted into the core. The ORR shim rods are withdrawn as a gang to compensate for fuel consumption; so the correction was made on all shim rods containing fuel.

Fuel Distribution

During the course of depletion, the fuel is not burned evenly but acquires a nonuniform distribution, because more fuel is consumed near the longitudinal center of the element than near the ends. This distribution has been determined for the ORR fuel elements by measuring the fission product distribution along the element.

The following empirical relationships have been developed to describe the actual fuel distribution along the element:

$$U(L)_{200} = 8.33 - (0.01164 + 0.0067594L - 0.00026607L^2)\,(200 - W),$$
$$U(L)_{240} = 10.00 - (0.01164 + 0.0067594L - 0.00026607L^2)\,(240 - W),$$
$$U(L)_{141} = 5.88 - (0.03714 + 0.0037278L - 0.0002094L^2)\,(141 - W),$$

where $U(L)_{200}$, $U(L)_{240}$, and $U(L)_{141}$ refer to the concentration of fuel remaining at L (grams/linear inch) in elements which originally contained 200 and 240 g and in shim rods which originally contained 141 g of fuel, L is the distance from the top of the fuel section (in.), and W is the calculated weight of fuel in a partially depleted fuel element or shim rod (g).

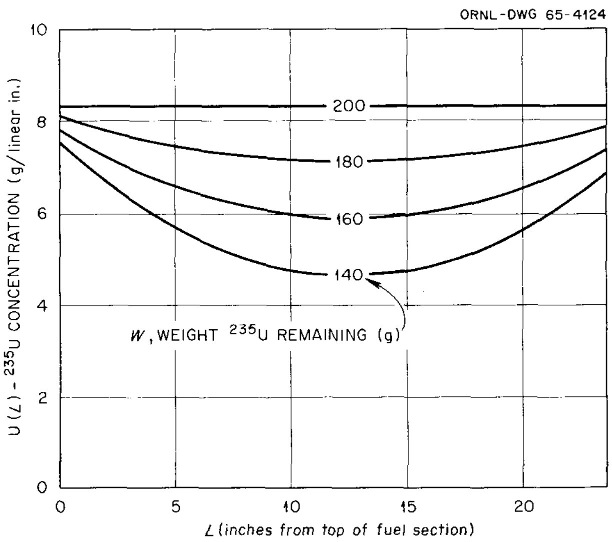


Fig. C.1. Distribution of Fuel Along the Length of a Partially Depleted ORR Fuel Element.

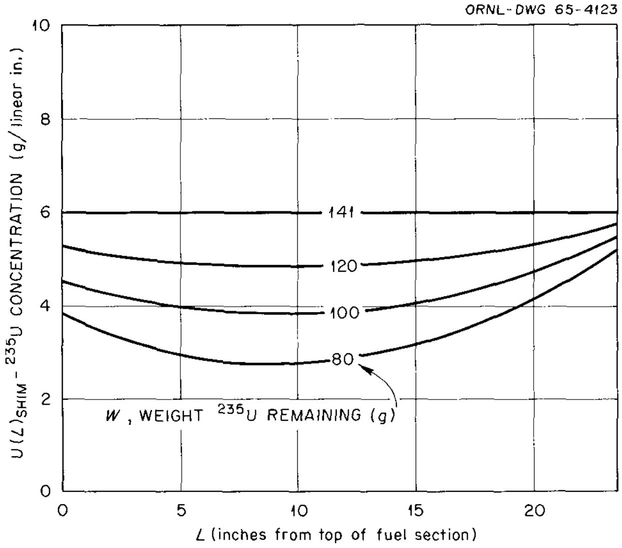


Fig. C.2. Distribution of Fuel Along the Length of a Partially Depleted ORR Shim-Rod Fuel Section.

Figures C.1 and C.2 show the fuel distribution for various burnups for two types of fuel sections under consideration.

Flux Distribution

Theoretical calculations have been used to establish the neutron flux distribution for unperturbed conditions and can at times estimate the perturbed flux distribution. However, the core arrangements for the ORR do not resemble a standard geometry and have perturbing absorbers in both the core and the reflector. For this reason, the flux distribution can only be determined by flux mapping.

A routine mapping of the ORR core is made about every six months or every time the core configuration is changed. These mappings are performed by inserting cobalt wire monitors along the longitudinal centers of the fuel elements and shim rods, raising the reactor to 30 kw for 30 min, shutting down, then removing and counting the monitors.

We do not feel that it is necessary to obtain absolute fluxes but rather to calculate flux ratios. The average neutron flux in the core $\bar{\Phi}_c$ can be calculated from the following relation:

$$\bar{\Phi}_c = \frac{\text{reactor power (w)} \times \text{fissions/wsec}}{N_u \times \sigma_f(\text{eff})},$$

where N_u is the number of ^{235}U atoms in the reactor and σ_f(eff) is the effective fission cross section for the ^{235}U in the reactor. For the ORR:

$$\bar{\Phi}_c = \frac{7036 \times 10^{14}}{\text{wt } ^{235}\text{U(g)}} \text{ neutrons cm}^{-2}\text{ sec}^{-1} .$$

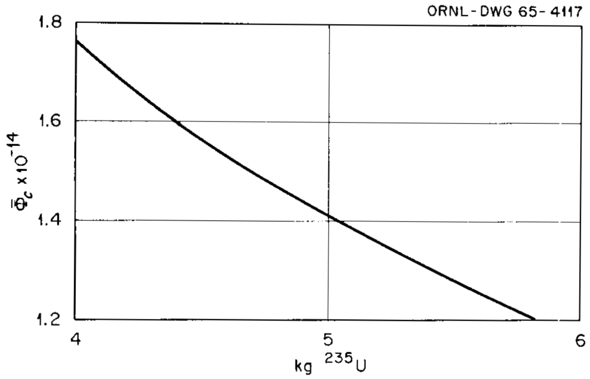


Fig. C.3. Average Thermal Neutron Flux in the ORR Core at 30 Mw vs Weight of Fuel in the Core.

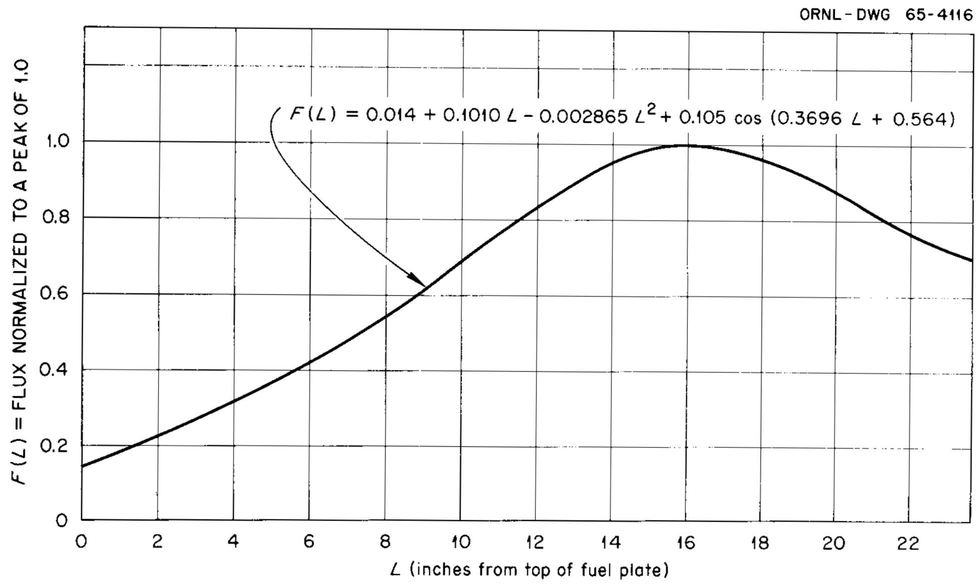


Fig. C.4. Flux Distribution Along the Length of an ORR Fuel Element Normalized to a Peak of 1.00.

Figure C.3 shows $\bar{\Phi}_c$ plotted against various core weights for the ORR. A typical traverse along an ORR fuel element is shown in Fig. C.4. This was obtained by cutting the cobalt wire into 1-in. segments normalized to a peak of 1. The traverse of an ORR shim-rod fuel section is shown in Fig. C.5.

The neutron flux is very much depressed in the upper portion of the core because of the poison sections of the shim rods. Although this distortion will tend to become less pronounced as burnup proceeds during a fuel cycle and the rods are withdrawn more, it will not completely disappear, even if the fuel cycle progresses until the shim rods are almost fully withdrawn, because the fuel distribution is also distorted.

With this background information, we can now proceed with the calculation of BUF-IV.

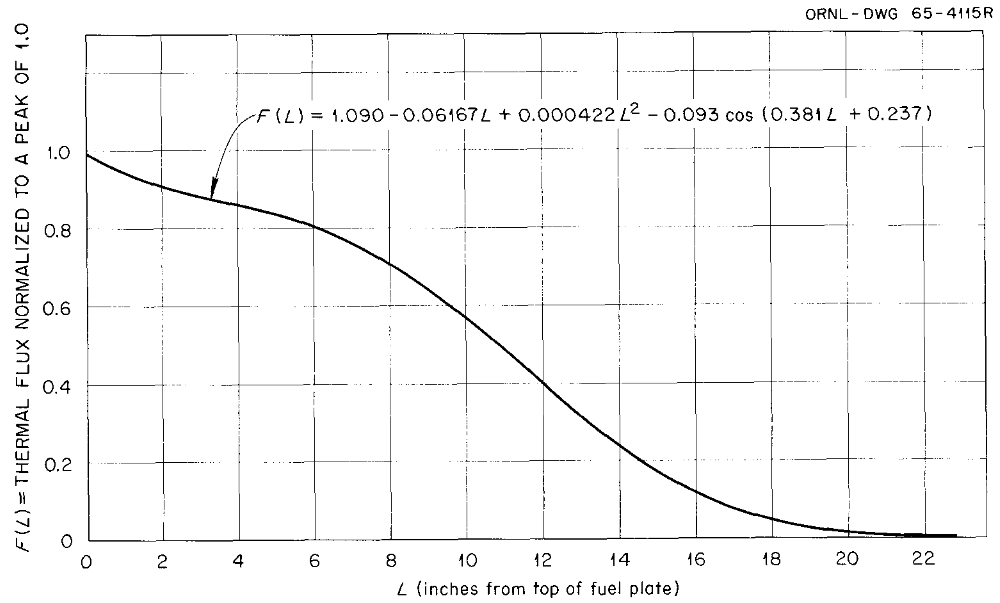


Fig. C.5. Flux Distribution Along the Length of an ORR Shim-Rod Fuel Section Normalized to a Peak of 1.00.

We will consider an element which originally contained 200 g and has been partially depleted to 180 g. This element has been in a core position in which $\overline{\Phi}_i/\overline{\Phi}_c = 1.000$. If BUF-III were calculated from this data, we would obtain

$$\text{BUF-III} \cong \frac{\overline{\Phi}_i}{\overline{\Phi}_c} \times \frac{M_i}{M_c} = \frac{180}{M_c},$$

but, if we consider the components as described above, we first look at the following formulas:

$$U(L)_{200} = 8.33 - (0.01164 + 0.0067594L - 0.00026607L^2)(200 - W).$$

For $W = 180$,

$$U(L)_{200} = 8.10 - 0.13519L + 0.0053214L^2$$

and

$$F(L) = 0.014 + 0.1010L - 0.002865L^2 + 0.105 \cos(0.3696L + 0.564).$$

For $\overline{\Phi}_i/\overline{\Phi}_c = 1.000$,

$$\frac{\overline{\Phi}_i(L)}{\overline{\Phi}_c} = 0.021 + 0.1507L - 0.004276L^2 + 0.157 \cos(0.3696L + 0.564).$$

Therefore, in terms of calculating BUF-III for $W = 180$,

$$\text{BUF-III} \cong \int_0^{24} \frac{\overline{\Phi}_i}{\overline{\Phi}_c}(L)\, dL \int_0^{24} \frac{U(L)_{200}}{24M_c}\, dL \cong \frac{181.06}{M_c}.$$

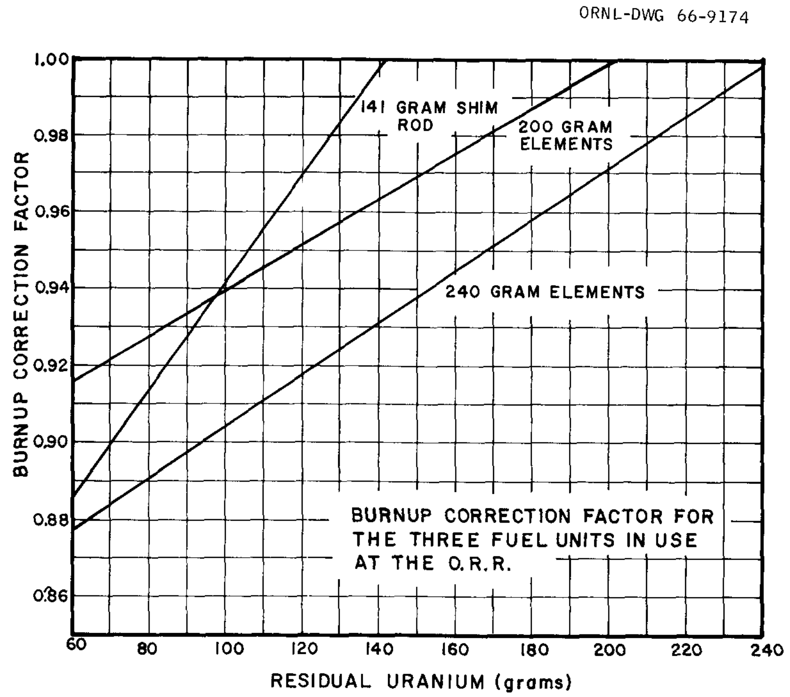


Fig. C.6. Burnup Correction Factor for the Three Fuel Units in Use at the ORR.

However, if the intervals are made smaller, say 1 in., we have

$$\text{BUF-IV} = \sum_{j=1}^{24} \int_{j-1}^{j} \frac{\bar{\Phi}_i}{\bar{\Phi}_c}(L)\, dL \int_{j-1}^{j} \frac{U(L)_{200}}{M_c}\, dL \underset{=}{\sim} \frac{177.51}{M_c} .$$

Dividing BUF-III by BUF-IV, we have 0.987. This means we are consuming only 98.7% of the amount calculated by BUF-III. Figure C.6 shows the correction factor which must be applied to BUF-III to obtain BUF-IV for the various elements under discussion.

Again, we can examine the hypothetical EBF core (Table C.4). Table C.5 compares the four burnup factors under discussion.

It is realized that the correction in going from BUF-III to BUF-IV is minor in the cases studied; however, the ORR is now being fueled with elements which originally contained 240 g of ^{235}U, and as depletion progresses the correction factor to be applied becomes more significant.

Comparison of Reactor Spent Fuel with Reported Recovery Results

Table C.6 shows the amounts of residual uranium in spent enriched fuel as calculated by ORNL, contrasted with the recovery results reported by the recovery facility. Also shown are the differences between our calculated weights and the reported recovery weights. The experiences shown cover a period from May 1953 through January 11, 1966 – a period of slightly over 12 years.

From May 1953 through 1957 spent fuel elements resulted from the operation of the LITR only. In 1958 operation of the ORR was begun, and the number of spent fuel elements requiring recovery increased substantially.

It is noted that of the 1321 spent fuel elements for which recovery results have been reported to date, the recovered material balance has been 99.988% of ORNL's total uranium calculation and 100.15% of ORNL's ^{235}U calculation.

Table C.4. Calculation of BUF-IV

Core Position	Weight of ^{235}U (g)	BUF-III	Correction Factor	BUF-III × Correction Factor	BUF-IV
A-1	200	0.06624	1.000	0.06624	0.06621
A-2	180	0.06814	0.987	0.06725	0.06722
A-3	160	0.06814	0.972	0.06623	0.06620
A-4	180	0.06384	0.987	0.06301	0.06298
A-5	200	0.06159	1.000	0.06159	0.06156
B-1	180	0.08431	0.987	0.08321	0.08317
B-2	120	0.05679[a]	0.970	0.05509	0.05506
B-3	160	0.09830	0.972	0.09555	0.09550
B-4	120	0.05111[a]	0.970	0.04958	0.04956
B-5	180	0.08088	0.987	0.07983	0.07979
C-1	200	0.06386	1.000	0.06386	0.06383
C-2	180	0.06599	0.987	0.06513	0.06510
C-3	160	0.06623	0.972	0.06438	0.06435
C-4	180	0.06171	0.987	0.06091	0.06088
C-5	200	0.05862	1.000	0.05862	0.05859
				1.00048	1.00000

[a]Not corrected by 0.854 as in previous BUF-III calculation.

Table C.5. Comparison of Burnup Factors

Core Position	BUF-I	BUF-II	BUF-III	BUF-IV
A-1	0.05553	0.05691	0.06624	0.06621
A-2	0.06347	0.06505	0.06814	0.06722
A-3	0.07141	0.07319	0.06814	0.06620
A-4	0.05947	0.06095	0.06384	0.06298
A-5	0.05133	0.05281	0.06159	0.06156
B-1	0.07853	0.08048	0.08431	0.08317
B-2	0.07933	0.08130	0.04850	0.05506
B-3	0.10299	0.10556	0.09830	0.09550
B-4	0.07141	0.04831	0.04365	0.04956
B-5	0.07533	0.07720	0.08088	0.07979
C-1	0.05353	0.05486	0.06386	0.06383
C-2	0.06147	0.06300	0.06599	0.06510
C-3	0.06940	0.07113	0.06623	0.06435
C-4	0.05747	0.05890	0.06171	0.06088
C-5	0.04913	0.05035	0.05862	0.05859

Table C.6. Recovery Results on Enriched Spent Fuel Elements

Weights in grams

Period of Shipments		Number of Spent Elements	ORNL Calculated Weight		Reported Recovery		Difference Between ORNL Weight and Reported Recovery Weight	
From	To		U	^{235}U	U	^{235}U	U	^{235}U
May 1953	December 1953	27	3,237	2,964	3,227	2,942	10	22
March 1954	August 1954	17	2,134	1,859	2,101	1,823	33	36
December 1954	June 1955	28	3,806	3,504	3,777	3,477	29	27
June 1955	April 1956	17	2,262	1,931	2,211	1,882	51	49
April 1956	November 1956	8	1,619	786	1,612	782	7	4
June 1958	December 1959	151	18,034	15,476	17,555	15,066	479	410
January 1960	January 1962	308	44,630	37,491	44,425	37,462	205	29
January 1962	September 1963	321	46,937	39,255	47,727	40,072	(790)	(817)
September 1963	November 1964	228	38,158	29,387	38,158	29,387	0	0
November 1964	December 1965	195	29,831	25,082	29,831	25,082	0	0
December 1965	January 1966	21	3,161	2,646	3,161	2,646	0	0
	Total	1,321	193,809	160,381	193,785	160,621	24	(240)

Appendix D

REACTOR CONTROLS INFORMATION FOR ORR SUPERVISORY PERSONNEL. SUPPLEMENTARY INFORMATION TO *ELEMENTARY REACTOR CONTROLS DIAGRAMS*

Compiled by R. A. Costner and T. P. Hamrick

This section consists of a listing of relays, recorder switches, and limit switches in the ORR control system. The relays are arranged in numerical order; the condition which must exist for the relay to be picked up is listed along with the relay location. The recorder switches are arranged first functionally and second numerically. The limit switches are arranged in numerical order.

REFERENCE DRAWINGS

RC2-1-3AA – Elementary Diagram Shim Rod Controls

RC2-1-3AB – Elementary Diagram Shim Rod Controls

RC2-1-3BA – Elementary Diagram Shim Rod Controls

RC2-1-3BB – Elementary Diagram Shim Rod Controls

RC2-1-3C – Elementary Diagram Motors & Synchros

RC2-10-2 – Safety Trouble Monitor

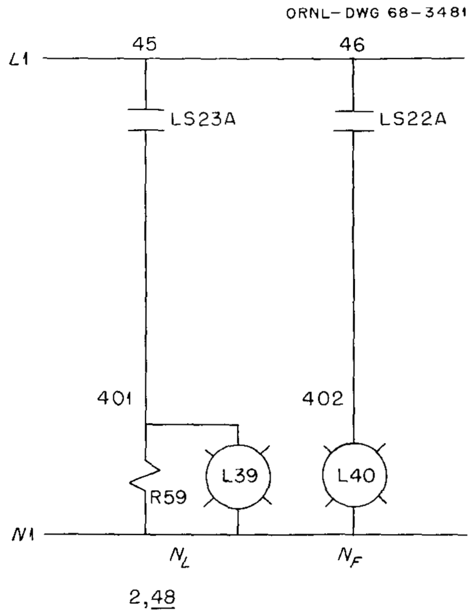
RC2-1-3E – Elementary Diagram Process Controls

RC2-1-3F – Elementary Diagram Process Controls

RC2-1-3G – Elementary Diagram Process Annunciators

RC2-1-3H – Typical Connection of Experiment to Reactor Controls

RC2-1-3I – Elementary Diagram Fission Chambers



Key to Drawings (Excerpt from Reference Drawing RC2-1-3BA)

L1, N1, 401, and 402	Denote wire numbers
45 and 46	Denote circuit numbers
LS22-A and LS23-A	Denote contact numbers (in this case, limit switches)
R59	Denotes the relay number
2, 48	Indicates that a normally open contact of R59 appears in circuit 2 and a normally closed contact of R59 appears in circuit 48 (normally closed is denoted by an underline)
N_L and N_F	Denote conditions (in this case the servo demand is at N_L or N_F)
L39 and L40	Are the pilot lights associated with the condition (N_L or N_F)

RELAYS

Table D.1 Numerical Listing

Relay	Circuit	Condition when Energized	Cabinet	Column	Row
R-1	4	Start request made	A-R	1	1
R-2	5	Reactor in start (sealed in)	A-R	3	1
R-3A	6	Reactor in run	A-R	4	1
R-3B	6	Reactor in run	A-R	5	1
R-4	7	Group rod withdrawal permitted	A-R	1	3
R-5	9	Group withdraw request	A-R	1	2
R-6	10	Individual shim rod movement requested	A-R	2	2
R-6X	10	Individual shim rod movement requested	X-F	3	6
R-7	12	No. 1 rod – withdraw	A-F	1	3
R-7X	15	No. 4 rod – withdraw	X-F	1	2
R-8	21	No. 1 rod – insert	A-F	1	6
R-8X	24	No. 4 rod – insert	X-F	1	5
R-9	13	No. 2 rod – withdraw	A-F	2	3
R-9X	16	No. 2 rod – withdraw	X-F	2	2
R-10	22	No. 2 rod – insert	A-F	2	6
R-10X	25	No. 5 rod – insert	X-F	2	5
R-11	14	No. 3 rod – withdraw	A-F	3	3
R-12	23	No. 3 rod – insert	A-F	3	6
R-13X	17	No. 6 rod – withdraw	A-R	2	5
R-14	27	No. 6 rod – insert	A-F	4	6
R-14X	27	No. 6 rod – insert	A-R	2	6

Table D.1 (continued)

Relay	Circuit	Condition when Energized	Cabinet	Column	Row
R-15	19	Insert 5 rods request	A-R	4	5
R-16	117	Servo rod above midpoint of L.R.C.[a]	A-F	6	4
R-17	20	Reverse request	A-R	1	6
R-17X	20		A-F	4	1
R-18A	30	Reactor in servo	A-F	5	5
R-18B	30		A-R	4	6
R-19					
R-20	17	Servo limit switches – withdraw	A-F	5	3
R-21	18	Preferred rod insert request	A-R	2	4
R-22	26	Servo limit switches – insert	A-F	5	6
R-23	47	Servo demand – raising	A-R	4	2
R-24	48	Servo demand – lowering (setback)	A-R	4	3
R-24X	48		A-R	3	2
R-24Y	49	Experiment setback request	X-F	4	6
R-25	187	No. 2 fission chamber – being withdrawn	A-R	6	3
R-26	186	No. 2 fission chamber – being inserted	A-R	5	3
R-27	52	No experiment – reverse request	A-F	6	2
R-27AG[b]	50		A-F	7	2
R-28	37	No slow scram	A-F	6	7
R-28X	40	Slow scram	A-F	6	1
R-29	58	$N > 1.8\ N_L$ on "selected" log N	A-R	1	4
R-30	28	Servo insert error	A-F	6	6
R-31AG[b]	53	No auxiliary reverse request (from reactor parameters)	A-F	6	3
R-32	206	No. 2 CRM on USE (i.e., not in CALIBRATE)	A-R	6	2
R-33	59	Period less than 30 sec on "selected" log N	A-F	6	5
R-34	74	No. 1 rod at upper limit	A-F	1	2
R-34X	92	No. 4 rod at upper limit	X-F	1	1
R-35	75	No. 1 rod at lower limit	A-F	1	7
R-35X	93	No. 4 rod at lower limit	X-F	1	6
R-36	76	No. 1 rod at seat	A-F	1	4
R-36X	94	No. 4 rod at seat	X-F	1	3
R-37	78	No. 1 rod has clutch	A-F	1	5
R-37X	96	No. 4 rod has clutch	X-F	1	4

Table D.1 (continued)

Relay	Circuit	Condition when Energized	Cabinet	Column	Row
R-38	80	No. 2 rod at upper limit	A-F	2	2
R-38X	98	No. 5 rod at upper limit	X-F	2	1
R-39	81	No. 2 rod at lower limit	A-F	2	7
R-39X	99	No. 5 rod at lower limit	X-F	2	6
R-40	82	No. 2 rod at seat	A-F	2	4
R-40X	100	No. 5 rod at seat	X-F	2	3
R-41	84	No. 2 rod has clutch	A-F	2	5
R-41X	102	No. 5 rod has clutch	X-F	2	4
R-42	86	No. 3 rod at upper limit	A-F	3	2
R-43	87	No. 3 rod at lower limit	A-F	3	7
R-44	88	No. 3 rod at seat	A-F	3	4
R-45	90	No. 3 rod has clutch	A-F	3	5
R-46	110	No. 6 rod at upper limit	A-F	4	2
R-47	111	No. 6 rod at lower limit	A-F	4	7
R-48	112	No. 6 rod at seat	A-F	4	4
R-49	114	No. 6 rod has clutch	A-F	4	5
R-50	116	Servo rod at withdrawal limit of L.R.C.[a]	A-F	5	4
R-51	118	Servo rod at or below the preferred limit switch of L.R.C.[a]	A-F	5	2
R-52	119	Servo rod at the reverse switch of L.R.C.[a] (Note: This is actually the "insert five rods" limit switch)	A-F	5	7
R-53	181	No. 2 fission chamber – withdraw limit	A-R	3	3
R-54	183	No. 2 fission chamber – insert limit	A-R	3	4
R-55	62	Period less than 5 sec on "selected" log N	A-R	1	5
R-56	61	$N > 0.6\, N_L$ on "selected" log N	A-F	2	1
R-57		Pool level low	3rd balcony, north JB-BL-44		
R-58		Pool level (low) alarm seal	3rd balcony, north JB-BL-44		
R-59	45	Servo demand – N_L	A-R	6	5
R-60	3	Instrument start seal	A-F	3	1
R-61	209	Period less than 5 sec on selected CRM	A-R	5	4
R-62	31	Fast period permit	A-R	3	6
R-63	1	Instrument start request	A-R	6	1
R-64	2	Reactor in instrument start	A-R	6	6
R-64A	2	Reactor in instrument start	A-R	4	4
R-64B	2	Reactor in instrument start	A-R	3	5

Table D.1 (continued)

Relay	Circuit	Condition when Energized	Cabinet	Column	Row
R-65	60	$N > 0.001\ N_L$ on "selected" log N (i.e., log N confidence)	A-R	2	3
R-66	210	Period greater than 30 sec on selected CRM	A-R	6	4
R-67	207	>20 cps on No. 2 CRM	A-R	5	5
R-68	208	>8000 cps on No. 2 CRM	A-R	5	6
R-69					
R-70	29	Not in bare midriff (i.e., log N confidence or below 8000 cps when requesting instrument start)	A-F	1	1
R-71	120	No. 1 rod is "preferred"	X-F	3	2
R-72	121	No. 2 rod is "preferred"	X-F	4	2
R-73	122	No. 3 rod is "preferred"	X-F	3	3
R-74	123	No. 4 rod is "preferred"	X-F	4	3
R-75	124	No. 5 rod is "preferred"	X-F	3	4
R-76	125	No. 6 rod is "preferred"	X-F	4	4
R-77	284	No "two safety troubles"	A-F	4	3
R-78	54	Reactor $\Delta P > 21.4$ psig	X-F	4	1
R-79	55	Primary flow > 17,000 gpm	X-F	3	1
R-80					
R-81					
R-82	51	Negative period on "selected" log N <100 sec	A-F	7	3
R-83	56	Cell ventilation flow < 2500 cfm (nominal)	X-F	5	1
R-84	57		X-F	6	1
R-85	188	No. 1 fission chamber – being withdrawn	X-F	5	2
R-86	189	No. 1 fission chamber – being inserted	X-F	6	2
R-87	191	No. 1 fission chamber – insert limit	X-F	3	2
R-88	193	No. 1 fission chamber – withdraw limit	X-F	4	2
R-89	200	>8000 cps on No. 1 CRM	X-F	5	3
R-90	201	>20 cps on No. 1 CRM	X-F	6	3
R-91	202	Active fission chamber not in motion	A-R	5	2
R-92	203	No. 2 fission chamber selected	X-F	5	4
R-93	204	No. 1 fission chamber selected	X-F	6	4
R-94	205	No. 1 CRM on USE (i.e., not in CALIBRATE)	X-F	5	5

Table D.1 (continued)

Relay	Circuit	Condition when Energized	Cabinet	Column	Row
		Special			
RF3B		Pool cooling flow > 550 gpm	JB-PX-No. 1		

[a]L.R.C. or "limited range of control" is that length of the rod allowed to the servo controls without operator action.

[b]AG – "agastat" – slow drop: 5 sec fast pickup.

ORNL-DWG 68-3482

ROW \ COLUMN	1	2	3	4	5	6	7
1	R-70	R-56	R-60	R-17X	SPARE	R-28X	
2	R-34	R-38	R-42	R-46	R-51	R-27	R-27AG
3	R-7	R-9	R-11	R-77	R-20	R-31AG	R-82
4	R-36	R-40	R-44	R-48	R-50	R-16	
5	R-37	R-41	R-45	R-49	R-18A	R-33	
6	R-8	R-10	R-12	R-14	R-22	R-30	
7	R-35	R-39	R-43	R-47	R-52	R-28	

Fig. D.1. Relay Cabinet "A" Front.

ORNL-DWG 68-3483

ROW \ COLUMN	1	2	3	4	5	6
1	R-1	T-1	R-2	R-3A	R-3B	R-63
2	R-5	R-6	R-24X	R-23	R-91	R-32
3	R-4	R-65	R-53	R-24	R-26	R-25
4	R-29	R-21	R-54	R-64A	R-61	R-66
5	R-55	R-13X	R-64B	R-15	R-67	R-59
6	R-17	R-14X	R-62	R-18B	R-68	R-64

Fig. D.2. Relay Cabinet "A" Rear.

ORNL-DWG 68-3484

ROW \ COLUMN	1	2	3	4	5	6
1	R-34X	R-38X	R-79	R-78	R-83	R-84
2	R-7X	R-9X	R-87 R-71	R-88 R-72	R-85	R-86
3	R-36X	R-40X	R-73	R-74	R-89	R-90
4	R-37X	R-41X	R-75	R-76	R-92	R-93
5	R-8X	R-10X			R-94	
6	R-35X	R-39X	R-6X	R-24Y		

Fig. D.3. Relay Cabinet "X" Front.

RECORDER SWITCHES

Table D.2 Functional Listing of ORR Recorder Switches

Instrument or Process	Closed Condition	Associated Circuit	Channel No. 1[a]	Channel No. 2	Channel No. 3
	Nuclear Instrumentation Recorders				
Count rate					
Meter	In use	205, 206	RS-100	RS-39	
Recorder	>1.5 cps	11	RS-95	RS-8	
	>20 cps	201, 207	RS-93	RS-41	
	>100 cps	188, 187	RS-92	RS-40	
	>8000 cps	200, 208	RS-94	RS-9	
Period rec.	>30 sec	210	RS-98	RS-37	
	>20 sec	11	RS-99	RS-36	
	>10 sec	11	RS-97	RS-43	
	>5 sec	209	RS-96	RS-38	
Log N					
Recorder	$>0.001\ N_L$	60	RS-4	RS-57	
	$>0.33\ N_L$	6	RS-5	RS-58	
	$>0.6\ N_L$	61	RS-6	RS-59	
	$>1.8\ N_L$	58	RS-7	RS-60	
Period rec.	Negative period $<\lvert 100 \text{ sec}\rvert$	51	RS-33	RS-65	
	<30 sec	59	RS-1	RS-61	
	>20 sec	11	RS-2	RS-62	
	>10 sec	11	RS-42	RS-64	
	<5 sec	62	RS-3	RS-63	
Servo amplifier	Withdraw error	17	RS-10		
	Insert error	27	RS-11		
Safety recorder	$>0.6\ N_F$	48	RS-67	RS-68	RS-69
	$<1.1\ N_F$	141	RS-12	RS-13	RS-14
	$>1.1\ N_F$	48	RS-30	RS-31	RS-32
	$>1.2\ N_F$	20	RS-44	RS-45	RS-46
Gamma chamber rec.	<147 on 0–150 scale	41	RS-26	RS-27	
Trouble monitor	No log N trouble on selected channel	60	RS-34		
	Process Instrumentation Recorders				
Flow					
Shutdown					
FX1B-1	>1200 gpm	AP-20	RS-25		
Primary					
FX1A-2	<90% main flow		RS-16		
FX1A-1	>17,000 gpm	55	RS-15		
FX1A-3	>14,000 gpm	37	RS-17		
FX1A-4	<14,000 gpm	40	RX-17X		

Table D.2 (continued)

Instrument or Process	Closed Condition	Associated Circuit	Channel No. 1[a]	Channel No. 2	Channel No. 3
Facility cooling					
North					
FS302A2, B2	<71 gpm	48	RS-78	RS-80	
FS302A3, B3	>69 gpm	53	RS-79	RS-81	
South (5 ft plug and annulus)					
FS731A2, B2	<61 gpm	48	RS-70	RS-72	
FS731A3, B3	>59 gpm	53	RS-71	RS-73	
South (24-in. annulus)					
FS751A2, B2	<51 gpm	48	RS-74	RS-76	
FS751A3, B3	>49 gpm	53	RS-75	RS-77	
Pressure					
ΔP					
PdX55C	>15.3 psig	37	RS-24		
PdX55B	<15.3 psig	40	RS-24X		
PdX55A	>21.4 psig	54	RS-66		
Temperature					
ΔT	<13.0°F	AP-13	RS-18	RS-47	
	>13.5°F	48	RS-19	RS-48	
	<13.5°F	53	RS-28	RS-51	
	<15.5°F	37	RS-20	RS-49	
	>15.5°F	40	RS-20X	RS-50	
Outlet temperature	<134°F	AP-15	RS-21	RS-52	
	<135°F	53	RS-29	RS-56	
	<135°F	48	RS-22	RS-53	
	>140°F	40	RS-23X	RS-55	
	<140°F	37	RS-23	RS-54	
Cell ventilation					
FS59A	>2500 cfm	56	RS-82		
FS59B	>2500 cfm	57	RS-83		
Pressurizable off-gas					
PS60B, 61B	<|–20| in. wg	48	RS-84	RS-86	
PS60C, 61C	>|–19| in. wg	53	RS-85	RS-87	
Normal off-gas					
PS63B, 64B	<|–20| in. wg	48	RS-88	RS-90	
PS63C, 64C	>|–19| in. wg	53	RS-89	RS-91	

[a]Where duplicate channels are not provided, switches are listed under channel No. 1.

Table D.3 Numerical Listing of ORR Recorder Switches

Recorder Switch	Circuit	Instrument	Closed Condition
RS-1	59	No. 1 log period rec.	<30 sec
RS-2	11	No. 1 log period rec.	>20 sec
RS-3	62	No. 1 log period rec.	<5 sec
RS-4	60	No. 1 log N rec.	>0.001 N_L
RS-5	6	No. 1 log N rec.	>0.33 N_L
RS-6	61	No. 1 log N rec.	>0.6 N_L
RS-7	58	No. 1 log N rec.	>1.8 N_L
RS-8	11	No. 2 count rate rec.	>1.5 cps
RS-9	208	No. 2 count rate rec.	>8000 cps
RS-10	17	Servo amplifier	Withdraw error
RS-11	27	Servo amplifier	Insert error
RS-12	141	No. 1 safety rec.	<1.1 N_F
RS-13	141	No. 2 safety rec.	<1.1 N_F
RS-14	141	No. 3 safety rec.	<1.1 N_F
RS-15	55	Main flow: FX1A-1	>17,000 gpm
RS-16		Main flow: FX1A-2	<90% main flow (not in use)
RS-17	37	Main flow: FX1A-3	>14,000 gpm
RS-17X	40	Main flow: FX1A-4	<14,000 gpm
RS-18	AP-13	No. 1 ΔT: Td RX-1	<13.0°F
RS-19	48	No. 1 ΔT: Td RX-1	>13.5°F
RS-20	37	No. 1 ΔT: Td RX-1	<15.5°F
RS-20X	40	No. 1 ΔT: Td RX-1	>15.5°F
RS-21	AP-15	No. 1 outlet temp. rec.	<134°F
RS-22	48	No. 1 outlet temp. rec.	>135°F
RS-23	37	No. 1 outlet temp. rec.	<140°F
RS-23X	40	No. 1 outlet temp. rec.	>140°F
RS-24	37	Reactor ΔP: PdX55C	>15.3 psig
RS-24X	40	Reactor ΔP: PdX55B	<15.3 psig
RS-25	AP-20	Shutdown cooling: FX1B-1	>1200 gpm
RS-26	41	No. 1 γ rec.	<147
RS-27	41	No. 2 γ rec.	<147
RS-28	53	No. 1 ΔT rec.	<13.5°F
RS-29	53	No. 1 outlet temp. rec.	<135°F
RS-30	48	No. 1 safety rec.	>1.1 N_F
RS-31	48	No. 2 safety rec.	>1.1 N_F
RS-32	48	No. 3 safety rec.	>1.1 N_F
RS-33	51	No. 1 log period rec.	Negative $\tau < \lvert 100 \text{ sec} \rvert$
RS-34	60	Trouble monitor	No "log N trouble"
RS-35		Log N confidence	Function superseded by RS-4 and RS-57
RS-36	11	No. 2 C.R. period rec.	>20 sec
RS-37	210	No. 2 C.R. period rec.	>30 sec
RS-38	209	No. 2 C.R. period rec.	>5 sec
RS-39	206	No. 2 C.R. meter	In "use"
RS-40	187	No. 2 C.R. rec.	>100 cps
RS-41	207	No. 2 C.R. rec.	>20 cps

Table D.3 (continued)

Recorder Switch	Circuit	Instrument	Closed Condition
RS-42	11	No. 1 log period rec.	>10 sec
RS-43	11	No. 2 C.R. period rec.	>10 sec
RS-44	20	No. 1 safety rec.	$>1.2\ N_F$
RS-45	20	No. 2 safety rec.	$>1.2\ N_F$
RS-46	20	No. 3 safety rec.	$>1.2\ N_F$
RS-47	AP-13	No. 2 ΔT rec.	$<13.0^{\circ}$F
RS-48	48	No. 2 ΔT rec.	$>13.5^{\circ}$F
RS-49	37	No. 2 ΔT rec.	$<15.5^{\circ}$F
RS-50	40	No. 2 ΔT rec.	$>15.5^{\circ}$F
RS-51	53	No. 2 ΔT rec.	$<13.5^{\circ}$F
RS-52	AP-15	No. 2 outlet temp. rec.	$<134^{\circ}$F
RS-53	48	No. 2 outlet temp. rec.	$>135^{\circ}$F
RS-54	37	No. 2 outlet temp. rec.	$<140^{\circ}$F
RS-55	40	No. 2 outlet temp. rec.	$>140^{\circ}$F
RS-56	53	No. 2 outlet temp. rec.	$<135^{\circ}$F
RS-57	60	No. 2 log N rec	$>0.001\ N_L$
RS-58	6	No. 2 log N rec.	$>0.33\ N_L$
RS-59	61	No. 2 log N rec.	$>0.6\ N_L$
RS-60	58	No. 2 log n rec.	$>1.8\ N_L$
RS-61	59	No. 2 log period rec.	<30 sec
RS-62	11	No. 2 log period rec.	>20 sec
RS-63	62	No. 2 log period rec.	<5 sec
RS-64	11	No. 2 log period rec.	>10 sec
RS-65	51	No. 2 log period rec.	Negative $\tau < \|100\ \text{sec}\|$
RS-66	54	Reactor ΔP: PdX55A	>21.4 psig
RS-67	48	No. 1 safety rec.	$>0.6\ N_F$
RS-68	48	No. 2 safety rec.	$>0.6\ N_F$
RS-69	48	No. 3 safety rec.	$>0.6\ N_F$
RS-70	48	South facility, 5-ft plug and annulus	<61 gpm
RS-71	53		>59 gpm
RS-72	48		<61 gpm
RS-73	53		>59 gpm
RS-74	48	South facility, 24-in. annulus	<51 gpm
RS-75	53		>49 gpm
RS-76	48		<51 gpm
RS-77	53		>49 gpm
RS-78	48	North facility	<71 gpm
RS-79	53	North facility	>69 gpm
RS-80	48	North facility	<71 gpm
RS-81	53	North facility	>69 gpm
RS-82	56	Cell ventilation	>2500 cfm
RS-83	57	Cell ventilation	>2500 cfm
RS-84	48	Pressurizable off-gas	$<\|-20\|$ in. wg
RS-85	53	Pressurizable off-gas	$>\|-19\|$ in. wg
RS-86	48	Pressurizable off-gas	$<\|-20\|$ in. wg
RS-87	53	Pressurizable off-gas	$>\|-19\|$ in. wg
RS-88	48	Normal off-gas	$<\|-20\|$ in. wg

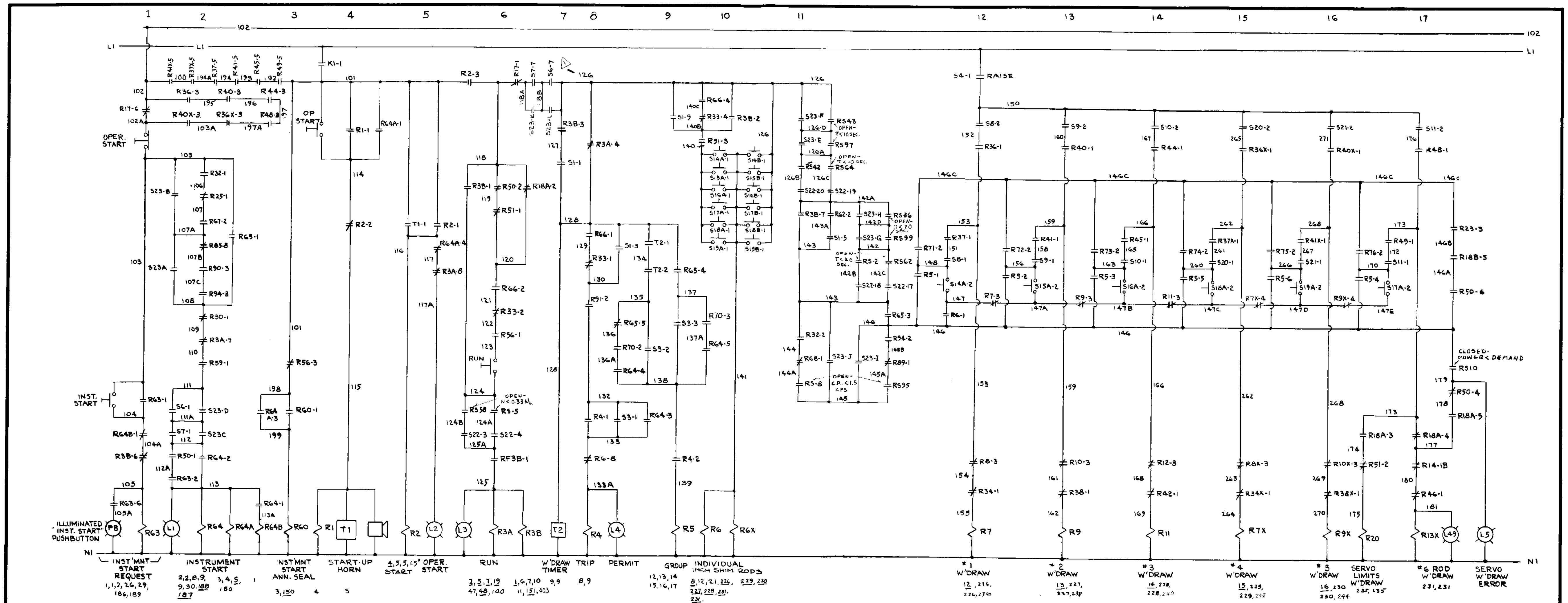
Table D.3 (continued)

Recorder Switch	Circuit	Instrument	Closed Condition
RS-89	53	Normal off-gas	$>\|-19\|$ in. wg
RS-90	48	Normal off-gas	$<\|-20\|$ in. wg
RS-91	53	Normal off-gas	$>\|-19\|$ in. wg
RS-92	188	No. 1 C.R. rec.	>100 cps
RS-93	201	No. 1 C.R. rec.	>20 cps
RS-94	200	No. 1 C.R. rec.	>8000 cps
RS-95	11	No. 1 C.R. rec.	>1.5 cps
RS-96	209	No. 1 C.R. period rec.	>5 sec
RS-97	11	No. 1 C.R. period rec.	>10 sec
RS-98	210	No. 1 C.R. period rec.	>30 sec
RS-99	11	No. 1 C.R. period rec.	>20 sec
RS-100	205	No. 1 C.R. meter	In "use"

LIMIT SWITCHES

Table D.4 Numerical Listing of ORR Limit Switches

Limit Switch	Circuit	Closed Condition
LS-1	74	No. 1 shim rod at upper limit
LS-1L1	191	No. 1 fission chamber *not* at insert limit
LS-1U1	193	No. 1 fission chamber *not* at withdraw limit
LS-2	75	No. 1 shim rod at lower limit
LS-3-1 & 2	76–77	No. 1 shim rod at seat
LS-4	78	No. 1 shim rod is "clutched"
LS-5	80	No. 2 shim rod at upper limit
LS-6	81	No. 2 shim rod at lower limit
LS-7-1 & 2	82–83	No. 2 shim rod at seat
LS-8	84	No. 2 shim rod is "clutched"
LS-9	86	No. 3 shim rod at upper limit
LS-10	87	No. 3 shim rod at lower limit
LS-11-1 & 2	88–89	No. 3 shim rod at seat
LS-12	90	No. 3 shim rod is "clutched"
LS-13	110	No. 6 shim rod at upper limit
LS-14	111	No. 6 shim rod at lower limit
LS-15-1 & 2	112–113	No. 6 shim rod at seat
LS-16	114	No. 6 shim rod is "clutched"
LS-17	116	Servo at withdraw limit
LS-18	118	Servo below preferred limit
LS-19	119	Servo at "insert 5 rods" limit
LS-20	181	No. 2 fission chamber *not* at withdraw limit
LS-21	183	No. 2 fission chamber *not* at insert limit
LS-22 A & B	46–47	Servo demand at N_F (A closed, B open)
LS-23 A & B	45	Servo demand at N_L (A closed, B open)
LS-24	117	Servo above middle limit
LS-25		Not used
LS-26		No. 1 pool level low (reactor pool)
LS-27		No. 2 pool level low (center pool)
LS-28		No. 3 pool level low (west pool)
LS-29	92	No. 4 shim rod at upper limit
LS-30	93	No. 4 shim rod at lower limit
LS-31-1 & 2	94–95	No. 4 shim rod at seat
LS-32	96	No. 4 shim rod is "clutched"
LS-33	98	No. 5 shim rod at upper limit
LS-34	99	No. 5 shim rod at lower limit
LS-35-1 & 2	100–101	No. 5 shim rod at seat
LS-36	102	No. 5 shim rod is "clutched"
LS-37	48	Servo demand > 500 kw (jumpered out)



S4 - RAISE-TEST

CONTACTS		POSITION TEST —	NORM —	RAISE —	LOC. CKT. No.
1 2	1			X	12
	2	X	X		37
3 4	3		X	X	461 (ANNUN)
	4	X			20
5 6	5	X			37
	6			X	40
7 8	7	X	X		—
	8		X	X	40
9 10	9	X			—
	10	X			—

S1 - REVERSE - BY-PASS

CONTACTS		POSITION REV.	NEUT. SR←	BY-PASS O	LOC. CKT. No.
1 2	1			X	7
	2	X			20
3 4	3			X	8
	4				—
5 6	5			X	11
	6				—
7 8	7				—
	8				—
9 10	9			X	10
	10				—

K1 - OPERATE SWITCH

CONTACTS		POSITION OP. —	OFF —	LOC. CKT. NO.
1 2	1	X		4
	2	X		41
3 4	3	X		41
	4			—

S3 GROUP WITHDRAW SW.

CONTACTS		POSITION HIGH O→	NEUT. SR	INTER.	LOC CKT. NO.
1 2	1		X		8
	2			X	9
3 4	3	X			9
	4				

S6 & S7 - FISSION CHAMBER

CONTACTS		POSITION W'DRAW O→	NEUT. SR	INSERT ←O	AUTO (PULL)	MAN. (PUSH)	S-7 CKT. NO.	S-6 CKT No.
1 2	1				X		2	2
	2				X		188	187
3 4	3					X	190	185
	4					X	—	—
5 6	5				X		189	186
	6				X		195	180
7 8	7		X				6	6
	8	X					188	187
9 10	9						—	—
	10			X			189	186

S-14, 15, 16, 17, 18 & 19
INDIVID. ROD MOTION
"A" "B"
W'DRAW INSERT

S23 - COUNTING CHANNEL SELECTOR

CONTACTS		POSITION CH#2 —	BOTH —	CH#1 —	LOC. CKT. NO.
A B	A	X			2
	B			X	2
C D	C	X			2
	D			X	2
E F	E	X			11
	F			X	11
G H	G	X			11
	H			X	11
I J	I	X			11
	J			X	11
K L	K	X			7
	L			X	7
M N	M	X			209
	N			X	209
O P	O	X			210
	P			X	210
Q R	Q	X			—
	R			X	—
S T	S	X			203
	T			X	204
U V	U	X			28
	V			X	28
W X	W		X	X	(PILTS)
	X	X	X		(PILTS)

S22 - LOG N CHANNEL SEL. SW.

CONTACTS		POSITION CH#2 —	CH#1 —	LOC CKT NO.
1 2	1	X		171
	2		X	170
3 4	3	X		6
	4		X	6
5 6	5	X		60
	6		X	60
7 8	7	X		61
	8		X	61
9 10	9	X		58
	10		X	58
11 12	11	X		59
	12		X	59
13 14	13	X		62
	14		X	62
15 16	15	X		51
	16		X	51
17 18	17	X		11
	18		X	11
19 20	19	X		11
	20		X	11
21 22	21	X		—
	22		X	—
23 24	23	X		—
	24		X	—

S-8, 9, 10, 11, 20 & 21 SHIM ROD MODE

CONTACTS		POSITION PREF —	NORM —	BLOCK —	OFF —	RAISE —	S-8 CKT. NO.	S-9 CKT. NO.	S-10 CKT. NO.	S-11 CKT. NO.	S-20 CKT. NO.	S-21 CKT. NO.
1 2	1	X	X				12	13	14	17	15	16
	2					X	12	13	14	17	15	16
3 4	3	X					—	—	—	—	—	—
	4		X	X	X	X	—	—	—	—	—	—
5 6	5	X					21	22	23	25	24	25
	6		X	X	X	X	21	22	23	—	24	25
7 8	7	X	X	X			21	22	23	27	24	25
	8	X					120	121	122	125	123	124
9 10	9		X	X	X	X	121	121	122	—	123	124
	10						—	—	—	—	—	—

RS NO	INSTRUMENT	CKT.
2	#1 LOG N τ RECORDER	11
5	#1 LOG N LEVEL RCRDR	6
8	#2 LOG C.R. RECORDER	11
10	SERVO AMPLIFIER	17
36	#2 CRM τ RECORDER	11
42	#1 LOG N τ RECORDER	11
43	#2 CRM τ RECORDER	11
58	#2 LOG N RECORDER	6
62	#2 LOG N τ RECORDER	11
64	#2 LOG N τ RECORDER	11
95	#1 LOG CR RECORDER	11
97	#1 CRM τ RECORDER	11
99	#1 CRM τ RECORDER	11

RELAY NO	1	2	3A	3B	4	5	6	6X	7	7X	8	8X	9	9X	10	10X	11	12	14	17	18A	18B	20	23	25	30	32	33	34	34X	36	36X	37	37X	38	38X
CKT. NO.	4	5	6	6	8	9	10	10	12	15	21	24	13	16	22	25	14	23	27	20	30	30	17	47	187	28	206	59	74	92	76	94	78	96	80	98

RELAY NO	40	40X	41	41X	42	44	45	46	48	49	50	51	56	59	60	62	63	64	64A	64B	65	66	67	68	70	71	72	73	74	75	76	85	89	90	91	94
CKT. NO.	82	100	84	102	86	88	90	110	112	114	116	118	61	45	3	31	1	2	2	2	60	210	207	208	29	120	121	122	123	124	125	188	200	201	202	205

* See Dwg RC2-1-4B

NO.	RC2-1-3AA REVISIONS	DATE	APP'D	APPD
1	ADD #2 ORR MEMO #63	12-16-63		DSA

DRAWN	DATE	SUBMITTED	DATE	APPROVED	DATE
W.F.G	5/21/63				
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE
D.S.A	23 May 63			a.E.A.B.	7 June 63

Replaces RC2-1-3A

REFERENCE DRAWINGS	DWG. NO.
ELEMENTARY DIAGRAM (CONTINUED)	RC2-1-3BB
ELEMENTARY DIAGRAM (CONTINUED)	RC2-1-3BA
ELEMENTARY DIAGRAM (CONTINUED)	RC2-1-3AB
COUNTING CHANNEL CONTROL ELEMENTRY	RC2-1-3I

DESIGN RESPONSIBILITY W.F. MRUK

ORR BLDG. 3042

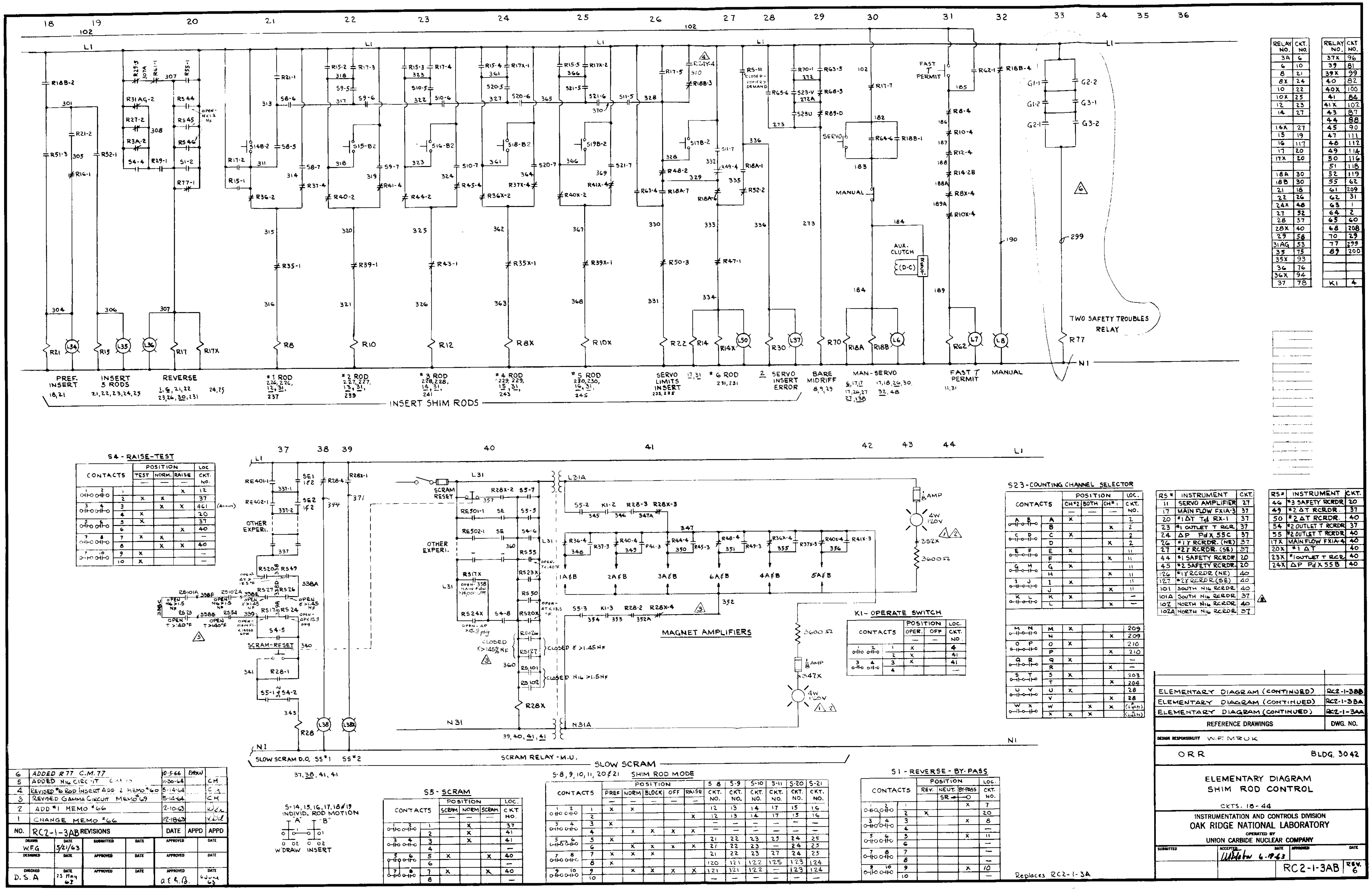
ELEMENTARY DIAGRAM
SHIM ROD CONTROLS

CKTS. 1-17

INSTRUMENTATION AND CONTROLS DIVISION
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY

SUBMITTED | ACCEPTED W.H. Tabor 6-18-63 DATE | APPROVED DATE

RC2·1·3AA REV 1



RELAY NO.	CKT. NO.
3A	6
6	10
8	21
8X	24
10	22
10X	25
12	23
14	27
14X	27
15	19
16	117
17	20
17X	20
18A	30
18B	30
21	18
22	26
24X	48
27	52
28	37
28X	40
29	58
31AG	53
35	75
35X	93
36	76
36X	94
37	78

RELAY NO.	CKT NO.
37X	96
39	81
39X	99
40	82
40X	100
41	84
41X	102
43	87
44	88
45	90
47	111
48	112
49	114
50	116
51	118
52	119
55	62
61	209
62	31
63	1
64	2
65	60
68	208
70	29
77	299
89	200
K1	4

RS*	INSTRUMENT	CKT.
11	SERVO AMPLIFIER	27
17	MAIN FLOW FXIA-3	37
20	*1 ΔT Td RX-1	37
23	*1 OUTLET T RCR.	37
24	ΔP PdX 55C	37
26	*1 γ RCRDR. (NE)	37
27	*2 γ RCRDR. (SE)	37
44	*1 SAFETY RCRDR.	20
45	*2 SAFETY RCRDR.	20
126	*1 γ RCRDR (NE)	40
127	*2 γ RCRDR (SE)	40
101	SOUTH N16 RCRDR	40
101A	SOUTH N16 RCRDR	37
102	NORTH N16 RCRDR	40
102A	NORTH N16 RCRDR	37

RS*	INSTRUMENT	CKT.
46	*3 SAFETY RCRDR	20
49	*2 ΔT RCRDR.	37
50	*2 ΔT RCRDR.	40
54	*2 OUTLET T RCRDR	37
55	*2 OUTLET T RCRDR	40
17X	MAIN FLOW FXIA-4	40
20X	*1 ΔT	40
23X	*1 OUTLET T RCR.	40
24X	ΔP PdX 55B	40

REFERENCE DRAWINGS	DWG. NO.
ELEMENTARY DIAGRAM (CONTINUED)	RC2-1-3BB
ELEMENTARY DIAGRAM (CONTINUED)	RC2-1-3BA
ELEMENTARY DIAGRAM (CONTINUED)	RC2-1-3AA

DESIGN RESPONSIBILITY W.F. MRUK

ORR BLDG. 3042

ELEMENTARY DIAGRAM
SHIM ROD CONTROL

CKTS. 18 - 44

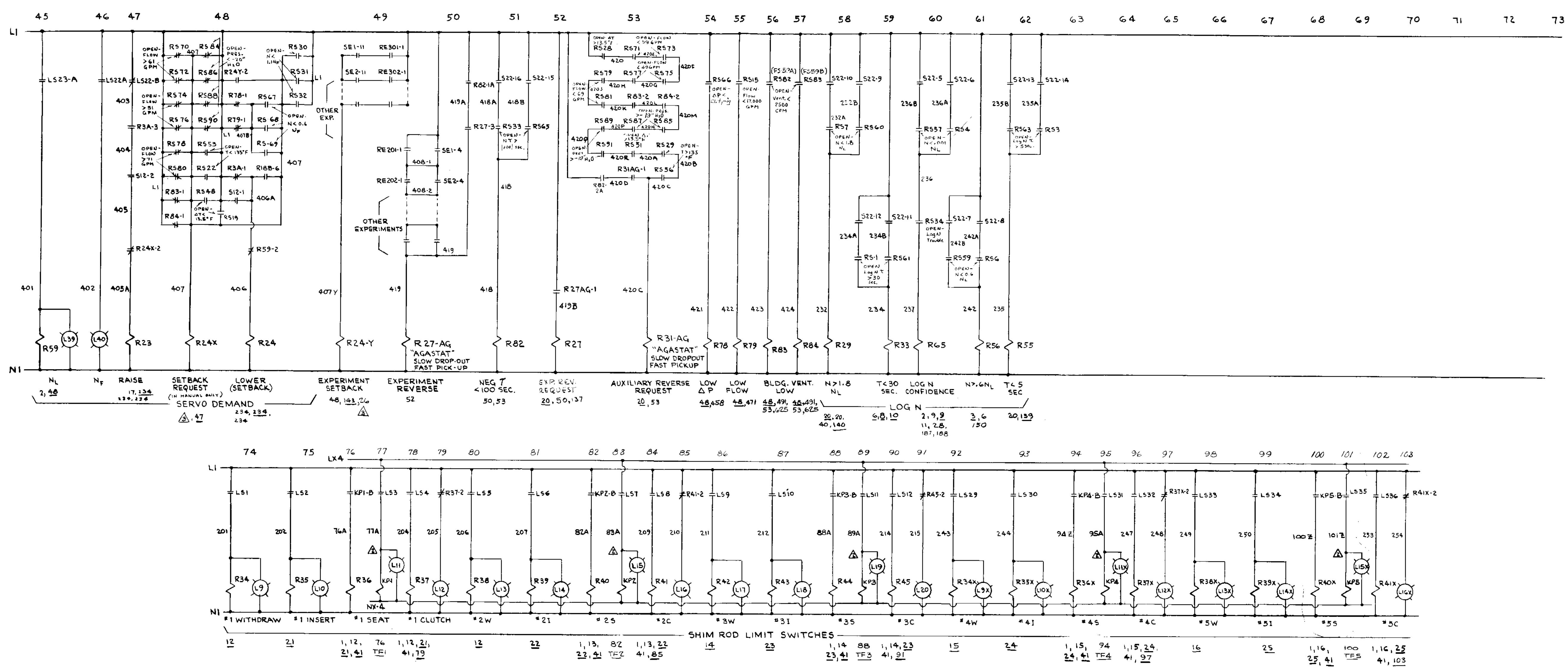
INSTRUMENTATION AND CONTROLS DIVISION
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY

SUBMITTED | ACCEPTED [signature] DATE 6-19-63 | APPROVED DATE

RC2-1-3AB REV. 6

NO.	RC2-1-3AB REVISIONS	DATE	APPD	APPD
6	ADDED R77 C.M. 77	10-5-66	DBW	
5	ADDED N16 CIRCUIT C.M. 70	11-30-64		CM
4	REVISED *6 ROD INSERT ADD 2 MEMO *60	5-14-64		CH
3	REVISED GAMMA CIRCUIT MEMO *69	5-14-64		CH
2	ADD *1 MEMO *66	2-10-63		Wdc
1	CHANGE MEMO *66	12-18-63		Wbd

DRAWN W.F.G.	DATE 5/21/63	SUBMITTED	DATE	APPROVED	DATE
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
CHECKED D.S.A	DATE 23 May 63	APPROVED	DATE	APPROVED a.E.G.B.	DATE 6 June 63



Note:
SEE DWG. RC2-13-3 FOR TF1 THRU TF6
(TIME OF FLIGHT CIRCUIT)

S12 - SERVO DEMAND

CONTACTS		POSITION RAISE	NEUT.	LOWER	LOC. CKT. NO
1 2	1			X	48
	2	X			47
3 4	3	X	X		—
	4				—
5 6	5	X			151
	6				—

S22 - LOG N CHANNEL SEL. SW.

CONTACTS		POSITION CH#2	CH#1	LOC. CKT. NO.
1 2	1	X		171
	2		X	170
3 4	3	X		6
	4		X	6
5 6	5	X		60
	6		X	60
7 8	7	X		61
	8		X	61
9 10	9	X		58
	10		X	58
11 12	11	X		59
	12		X	59
13 14	13	X		62
	14		X	62
15 16	15	X		51
	16		X	51
17 18	17	X		11
	18		X	11
19 20	19	X		11
	20		X	11
21 22	21	X		–
	22		X	–
23 24	23	X		–
	24		X	–

LS	CKT.
1	74
2	75
3	76/77
4	78
5	80
6	81
7	82/83
8	84
9	86
10	87
11	88/89
12	90
22	46/47
23	45
29	92
30	93
31	94/95
32	96
33	98
34	99
35	100/101
36	102
37	80

RELAY	CKT
3A	6
18A	30
18B	30
23	47
24	48
24X	48
24Y	49
27	52
27AG	50
29	58
31AG	53
33	59
34	74
34X	92
35	75
35X	93
36	76
36X	94
37	78
37X	96
38	78

RELAY	CKT.
38X	98
39	81
39X	99
40	82
40X	100
41	84
41X	102
42	86
43	87
44	88
45	90
55	62
56	61
59	45
65	60
78	54
79	55
82	51
83	56
84	57

RS#	INSTRUMENT	CKT
1	#1 LOG N T RECORDER	59
3	#1 LOG N T RECORDER	62
4	#1 LOG N RCDR	60
6	" " "	61
7	" " "	58
15	MAIN FLOW FXIA-1	55
19	#1 ΔT TdRX-1	48
22	#1 OUTLET T. RCDR.	48
28	#1 ΔT RCDR.	53
29	#1 OUTLET T. RCDR	53
30	#1 SAFETY RCDR.	48
31	#2 " "	48
32	#3 " "	48
33	#1 LOG N T RCDR.	51
34	TROUBLE MONITOR	60
48	#2 ΔT RCDR.	48
51	" " "	53
53	#2 OUTLET T. RCDR.	48
56	" " " "	53
57	#2 LOG N RCDR.	60
59	" " "	61
60	" " "	58
–	–	–
61	#2 LOG N T RCDR	59
63	" " " "	62
65	" " " "	51

RS#	INSTRUMENT	CKT
66	ΔP PdX55A	54
67	#1 SAFETY RCDR.	48
68	#2 " "	48
69	#3 " "	48
70	SOUTH FACILITY	48
71	" "	53
72	" "	48
73	" "	53
74	" "	48
75	" "	53
76	" "	48
77	" "	53
78	NORTH FACILITY	48
79	" "	53
80	" "	48
81	" "	53
82	CELL VENTILATION	56
83	" "	57
84	POG PRESSURE	48
85	" "	53
86	" "	48
87	" "	53
88	NOG PRESSURE	48
89	" "	53
90	" "	48
91	" "	53

REFERENCE DRAWINGS	DWG. NO.
MASTER DWG. LISTS	RC2-0-0
RELAY CAB. "X" WIRING FRONT PANEL	RC2-5-2C
RELAY CAB. WIRING FRONT PANEL	RC2-5-2A
VERT. PANELS	
CONSOLE WIRING-CENTER PANELS	RC2-2-2B
Continued on RC2-1-3AA, -3AB & -3BB	

NO.	RC2-1-3BA REVISIONS	DATE	APPD	APPD
3	CHANGE MEMO #60 ADD #2	6-15-64		
2	CHANGE MEMO #67	2-11-64		

DRAWN	DATE	SUBMITTED	DATE	APPROVED	DATE
W.F.G.	5-21-63				
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE
D.S.A	23MAY63			a.E.G.B.	6 June 63

DESIGN RESPONSIBILITY W.F. MRUK

ORR BLDG 3042

ELEMENTARY DIAGRAM
SHIM ROD CONTROLS

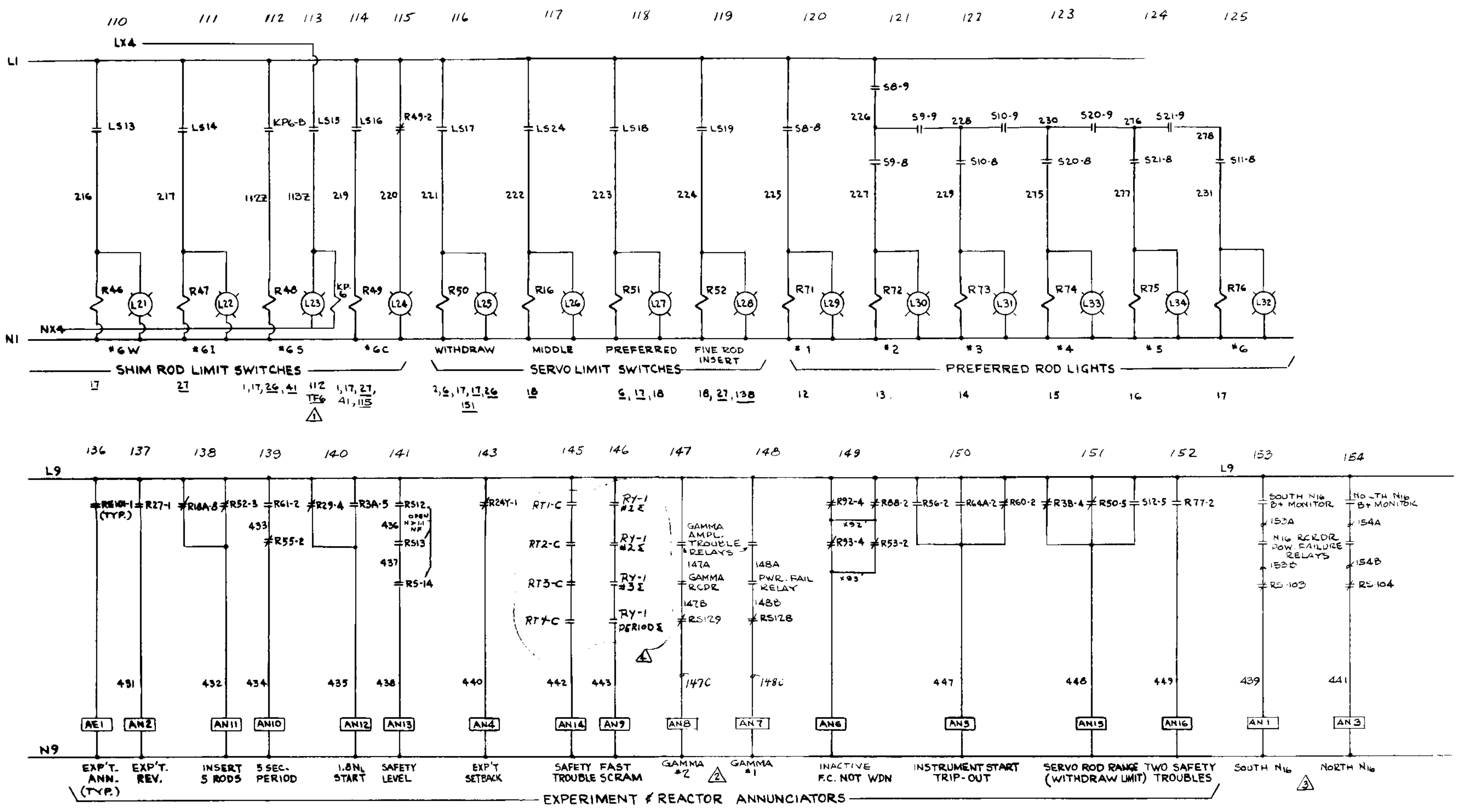
CKTS. 45-109

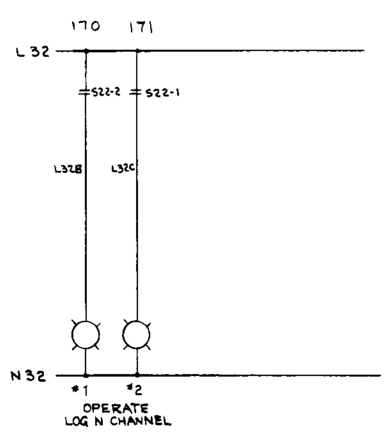
INSTRUMENTATION AND CONTROLS DIVISION
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY

SUBMITTED	ACCEPTED	DATE	APPROVED	DATE
	W.H. Tabor	6-18-63		

RC2-1-3BA REV. 3

Replaces RC2-1-3B





S12 - SERVO DEMAND

CONTACTS		POSITION RAISE	NEUT.	LOWER	LOC. CKT. NO
1 2	1			X	48
	2	X			47
3 4	3	X	X		–
	4				–
5 6	5	X			151
	6				–

S-8, 9, 10, 11, 20 & 21 SHIM ROD MODE

CONTACTS		POSITION PREF	NORM	BLOCK	OFF	RAISE	S-8 CKT. NO.	S-9 CKT. NO.	S-10 CKT. NO.	S-11 CKT. NO.	S-20 CKT. NO.	S-21 CKT. NO.
1 2	1	X	X				12	13	14	17	15	16
	2					X	12	13	14	17	15	16
3 4	3	X					–	–	–	–	–	–
	4		X	X	X	X	–	–	–	–	–	–
5 6	5	X					21	22	23	25	24	25
	6		X	X	X	X	21	22	23	–	24	25
7 8	7	X	X	X			21	22	23	27	24	25
	8	X					120	121	122	125	125	124
9 10	9		X	X	X	X	121	121	122	—	123	124
	10						–	–	–	–	–	–

S22 - LOG N CHANNEL SEL. SW.

CONTACTS		POSITION CH#2	CH#1	LOC. CKT. NO.
1 2	1	X		171
	2		X	170
3 4	3	X		6
	4		X	6
5 6	5	X		60
	6		X	60
7 8	7	X		61
	8		X	61
9 10	9	X		58
	10		X	58
11 12	11	X		59
	12		X	59
13 14	13	X		62
	14		X	62
15 16	15	X		51
	16		X	51
17 18	17	X		11
	18		X	11
19 20	19	X		11
	20		X	11
21 22	21	X		–
	22		X	–
23 24	23	X		–
	24		X	–

LS	CKT.
13	110
14	111
15	112/113
16	114
17	116
18	118
19	119
24	117

RS	INSTRUMENT	CKT.
12	#1 Safety RCDR	141
13	#2 " "	141
14	#3 " "	141
103	SOUTH N_{16} RCRDR	153
104	NORTH N_{16} RCRDR	154
128	#1 γ RCRDR (NE)	148
129	#2 γ RCRDR (SE)	147

RELAY	3A	3B	16	18A	24Y	27	29	46	47	48	49	50	51	52	53	56		60	61	64	71	72	73	74
CKT.	6	6	117	30	49	52	58	110	111	112	114	116	118	119	181	61		3	209	2	120	121	122	123
RELAY	75	76	77		88	92	93								"F" Series		"A" Series		"B,C,D,E"		"EE"			
CKT	124	125	284		193	203	204								ΣAmpl		Safety Trb. Mon.		Radiation Mons.		Exp. Mon.			

Replaces RC2-1-3B

NO.	RC2-1-3BB REVISIONS	DATE	APPD	APPD
4	CHANGE MEMO #77	6-30-67	WHM	
3	ADDED N_{16} CIRCUIT C.M. 70	11-23-64		CM
2	REVISED GAMMA CIRCUIT ADD. 6 MEMO #40	6-3-64		CM
1	CHANGE MEMO #67			

DRAWN	DATE	SUBMITTED	DATE	APPROVED	DATE
W.F.G.	5-21-63				
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE
ENGINEER	DATE	APPROVED	DATE	APPROVED	DATE
D.S.A	23 May 63			A.G.G.B.	5 June 63

REFERENCE DRAWINGS	DWG. NO.
MASTER DWG. LISTS	RC2-0-0
RELAY CAB "X" WIRING FRONT PANEL	RC2-5-2C
RELAY CAB. WIRING FRONT PANEL	RC2-5-2A
CONSOLE WIRING - CENTER PANELS	RC2-2-2B
Continued From RC2-1-3AA, -3AB & -3BA	

DESIGN RESPONSIBILITY W. H. MRUK

ORR BLDG. 3042

ELEMENTARY DIAGRAM
SHIM ROD CONTROLS

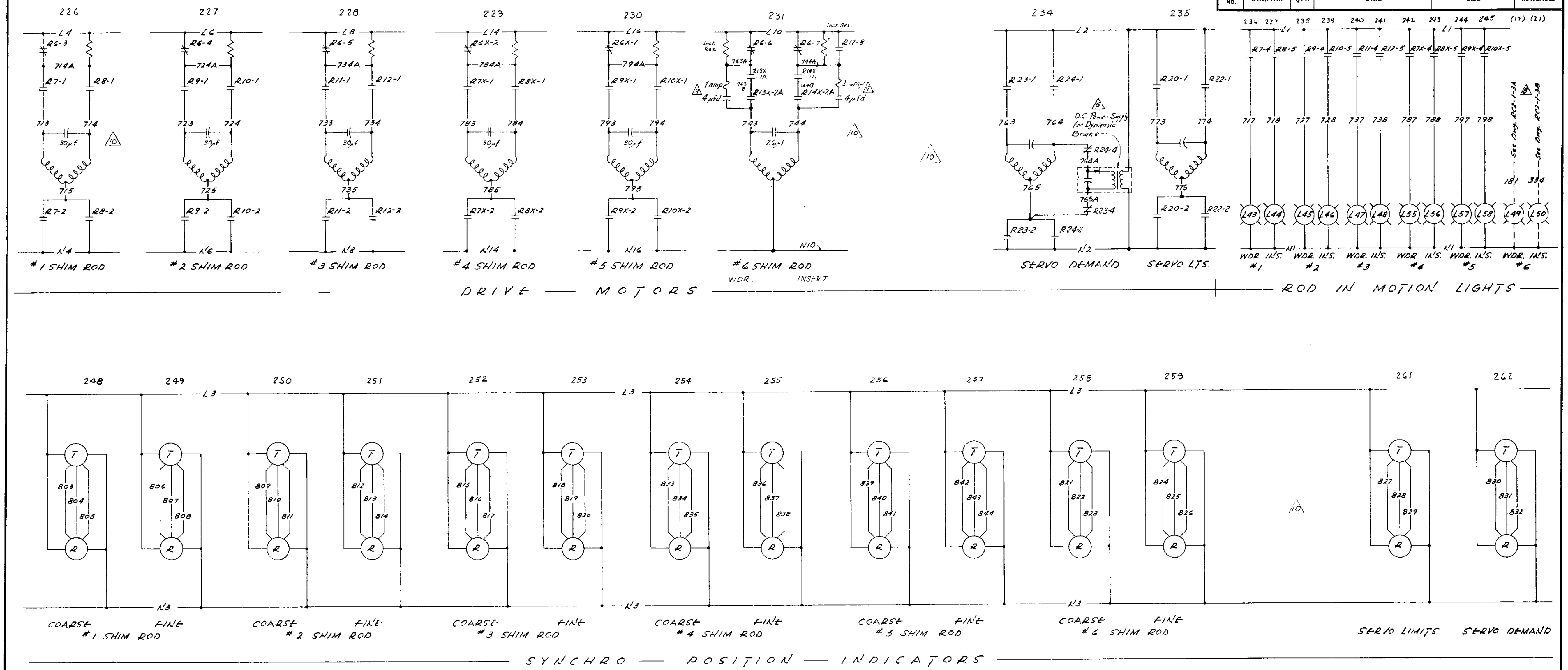
CKTS. 110-179

INSTRUMENTATION AND CONTROLS DIVISION
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY

SUBMITTED | ACCEPTED WHM 6-18-63 | APPROVED

RC2-1-3BB REV. 4

MATERIALS LIST					
ITEM NO.	DWG. NO.	QTY.	NAME	SIZE	MATERIAL



NO.	REVISIONS	DATE	APPR.	APPR.
11	Add Ckt. Nos. & Ref.	5 June 63		
10	Add 2nd Counting Channel Memo #63 Correct Motor Ckts	8 Mar. 63		
9	O.R.R. MEMO #26	3-28-60		
8	O.R.R. MEMO #24	3-16-60		
7	As wired.	4-2-58		
6	Added D.C. Brake To Fission Ch. Drive	11-13-57		
5	Added Dynamic Brake D.C. Power Supply	10-10-57		
4	General Revision	5-9-57		

DRAWING NO. RC2-1-3C

DRAWN	DATE	CHECKED	DATE	APPROVED	DATE
BISHOP	5-9-57				
DESIGNED	DATE	SUBMITTED	DATE	APPROVED	DATE

HEALTH PHYSICS	MEDICAL	OPERATIONS	SAFETY
FIRE PROTECTION	RESEARCH SHOP	MAINTENANCE	

Contact LS #	Device	Aspect
1	Shim Rod #1	Closed In Wdr. Limit
2	"	" " Ins. "
3	"	" " Seat
4	"	Closed When Clutch Is Connected
5	Shim Rod #2	Closed In Wdr. Limit
6	"	" " Ins. "
7	"	" " Seat
8	"	Closed When Clutch Is Connected
9	Shim Rod #3	Closed In Wdr. Limit
10	"	" " Ins. "
11	"	" " Seat
12	"	Closed When Clutch Is Connected
13	Shim Rod #6	Closed In Wdr. Limit
14	"	" " Ins. "

Contact LS #	Device	Aspect
15	Shim Rod #6	Closed In Seat
16	"	Closed When Clutch Is Connected
17	Servo	Closed In Wdr. Limit
18	"	Closed Below Pref Limit
19	"	Closed In Rev. Limit
20	Fission Ch. Drive	Closed in Wdr. Limit
21	"	" " Ins. "
22A & B	Servo Demand	A-Closed In N_F Limit - B Opens
23A & B	"	A-Closed In N_L Limit - B Opens
24	Servo	Closes Above Middle L.S.
25		
26	#1 Pool Level	Closes Low Level
27	#2 " "	" " "
28	#3 " "	" " "

Contact LS #	Device	Aspect
29	Shim Rod #4	Closed In Wdr. Limit
30	"	" " Ins. "
31	"	" " Seat
32	"	Closed When Clutch Is Connected
33	Shim Rod #5	Closed In Wdr. Limit
34	"	" " Ins. "
35	"	" " Seat
36	"	Closed When Clutch Is Connected
37	Servo Demand	Closed Demand > 500 KW

THIS DRAWING CLASSIFIED AS NOT CLASSIFIED
PER
SCALE:

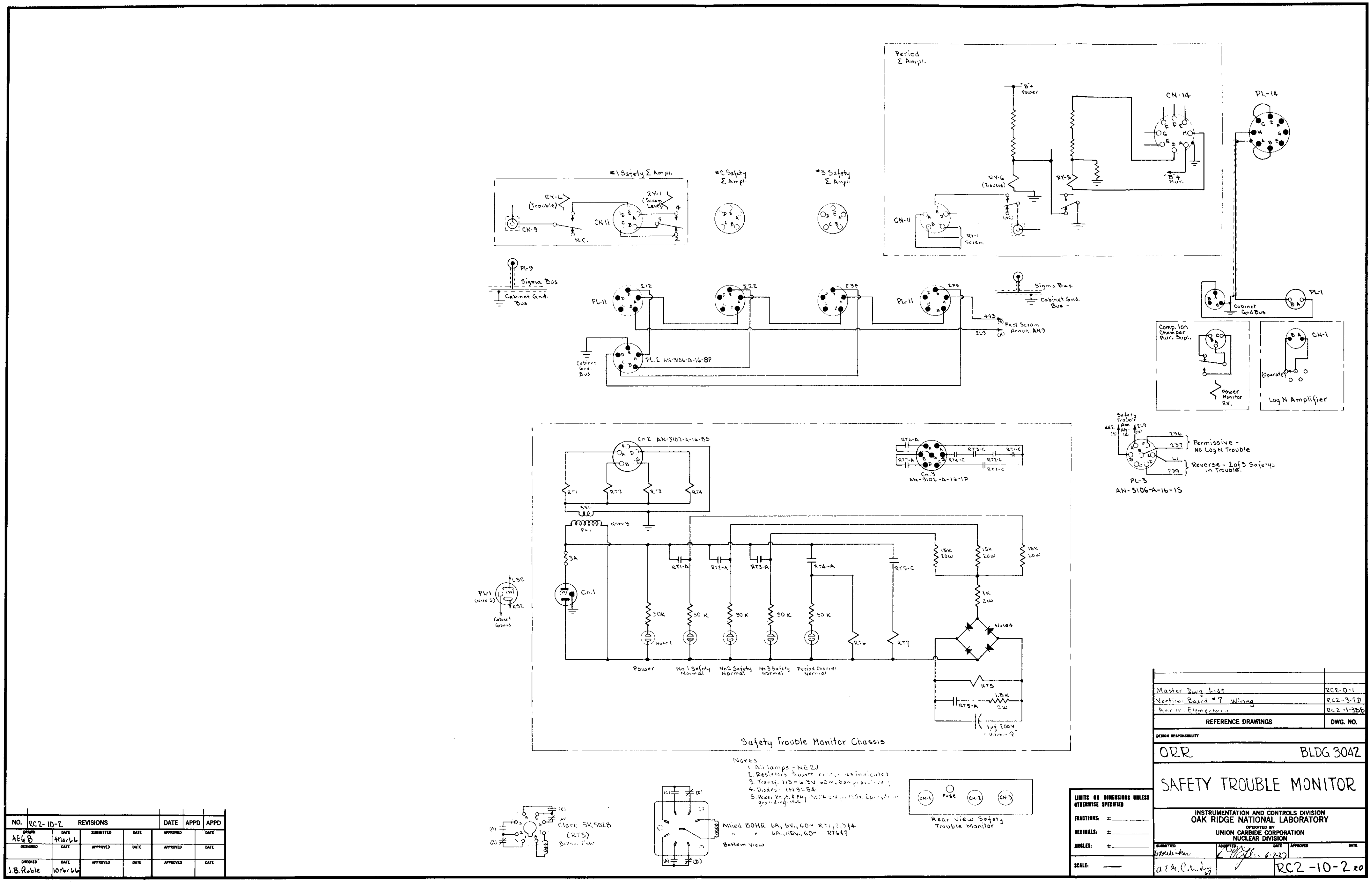
REFERENCE DRAWINGS	DWG. NO.
Control Elem. RC2-1-3AA, 3AB, 3BA, 3BB	
FISS. CH. DRIVE ELEM.	RC2-1-31

O.R.R. BLDG. 3042
ELEMENTARY DIAGRAM
MOTORS & SYNCHROS CKTS. 226-262

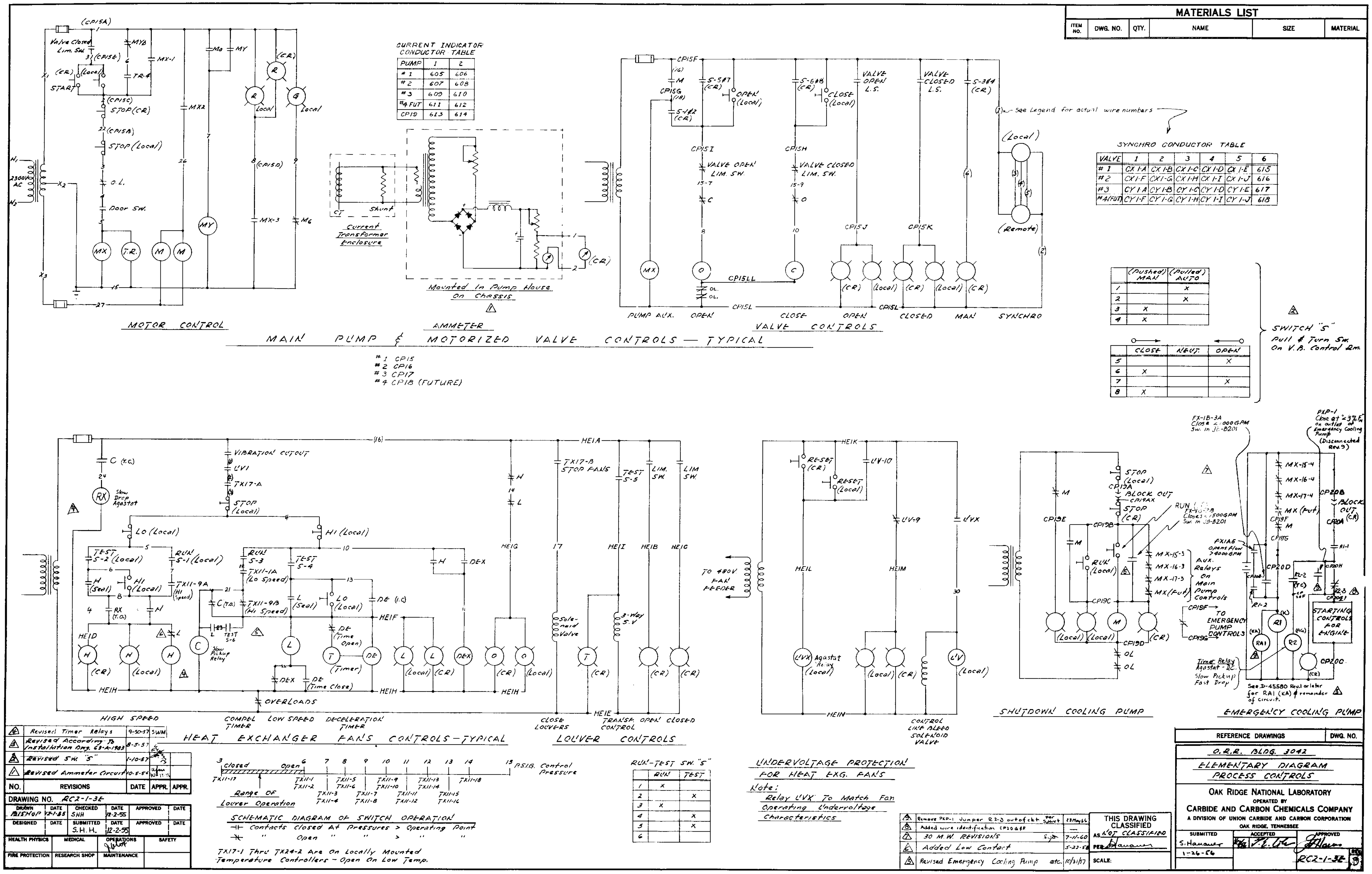
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY
A DIVISION OF UNION CARBIDE AND CARBON CORPORATION
OAK RIDGE, TENNESSEE

SUBMITTED 6-25-51 | ACCEPTED | APPROVED

RC2-1-3C REV 11



#1 Safety Σ Ampl.
#2 Safety Σ Ampl.
#3 Safety Σ Ampl.
Period Σ Ampl.
CN-14
PL-14
CN-11
CN-9
PL-9
Sigma Bus
Cabinet Gnd. Bus
PL-11
PL-2 AN-3106-A-16-8P
Fast Scram Annun. AN9
PL-1
CN-1
Comp. Ion Chamber Pwr. Supl.
Power Monitor RY.
Log N Amplifier
Cn 2 AN-3102-A-16-8S
Cn 3 AN-3102-A-16-1P
PL-3 AN-3106-A-16-1S
Permissive - No Log N Trouble
Reverse - 2 of 3 Safetys in Trouble
Power
No.1 Safety Normal
No.2 Safety Normal
No.3 Safety Normal
Period Channel Normal
Safety Trouble Monitor Chassis
Notes
1. All lamps - NE 2J
4. Diodes - 1N3254
Clare SK5028 (RT5) Bottom View
Allied BO4R 6A, 6V., 60~ RT1,2,3,4
Bottom View
Rear View Safety Trouble Monitor
NO. RC2-10-2 REVISIONS DATE APPD APPD
DRAWN AEGB 4 Mar 66
CHECKED J.B. Ruble 10 Mar 66
Master Dwg List RC2-0-1
Vertical Board #7 Wiring RC2-3-2D
REFERENCE DRAWINGS DWG. NO.
DESIGN RESPONSIBILITY
ORR BLDG 3042
SAFETY TROUBLE MONITOR
INSTRUMENTATION AND CONTROLS DIVISION
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE CORPORATION
NUCLEAR DIVISION
LIMITS ON DIMENSIONS UNLESS OTHERWISE SPECIFIED
FRACTIONS: ±
DECIMALS: ±
ANGLES: ±
SCALE:
SUBMITTED ACCEPTED DATE APPROVED DATE
RC2-10-2 R0



MATERIALS LIST

ITEM NO.	DWG. NO.	QTY.	NAME	SIZE	MATERIAL

CURRENT INDICATOR CONDUCTOR TABLE

PUMP	1	2
#1	605	606
#2	607	608
#3	609	610
#4 FUT	611	612
CP19	613	614

SYNCHRO CONDUCTOR TABLE

VALVE	1	2	3	4	5	6
#1	CX1-A	CX1-B	CX1-C	CX1-D	CX1-E	615
#2	CX1-F	CX1-G	CX1-H	CX1-I	CX1-J	616
#3	CY1-A	CY1-B	CY1-C	CY1-D	CY1-E	617
#4(FUT)	CY1-F	CY1-G	CY1-H	CY1-I	CY1-J	618

SWITCH "S"

	(Pushed) MAN	(Pulled) AUTO
1		X
2		X
3	X	
4	X	

	CLOSE	NEUT.	OPEN
5			X
6	X		
7			X
8	X		

SCHEMATIC DIAGRAM OF SWITCH OPERATION

Range of Louver Operation — 3 Closed ... Open 6 7 8 9 10 11 12 13 14 15 P.S.I.G. Control Pressure

⊣⊢ Contacts Closed At Pressures > Operating Point
⊬ " Open " " > " "

TX17-1 Thru TX24-2 Are On Locally Mounted Temperature Controllers – Open On Low Temp.

RUN-TEST SW. "S"

	RUN	TEST
1	X	
2		X
3	X	
4		X
5		X
6		

UNDERVOLTAGE PROTECTION FOR HEAT EXG. FANS

Note: Relay UVX To Match Fan Operating Undervoltage Characteristics

NO.	REVISIONS	DATE	APPR.	APPR.
4	Revised Timer Relays	9-30-57	SWM	
3	Revised According To Installation Dwg. 63-A-1983	8-5-57		
2	Revised Sw. "S"	1-10-57		
1	Revised Ammeter Circuit	10-5-56		

DRAWING NO. RC2-1-3E

DRAWN	DATE	CHECKED	DATE	APPROVED	DATE
BISHOP	12-1-55	SHH	12-2-55		
DESIGNED	DATE	SUBMITTED	DATE	APPROVED	DATE
		S.H.H.	12-2-55		

HEALTH PHYSICS	MEDICAL	OPERATIONS	SAFETY
FIRE PROTECTION	RESEARCH SHOP	MAINTENANCE	

9	Remove PKP-1 Jumper R2-3 out of ckt per S. Hurt	23 May 66
8	Added wire identification CP20 G & F	—
7	30 M.W. REVISIONS	7-11-60
6	Added Low Contact	5-22-58
5	Revised Emergency Cooling Pump etc.	10/31/57

THIS DRAWING CLASSIFIED AS NOT CLASSIFIED PER S. Hanauer
SCALE:

REFERENCE DRAWINGS	DWG. NO.

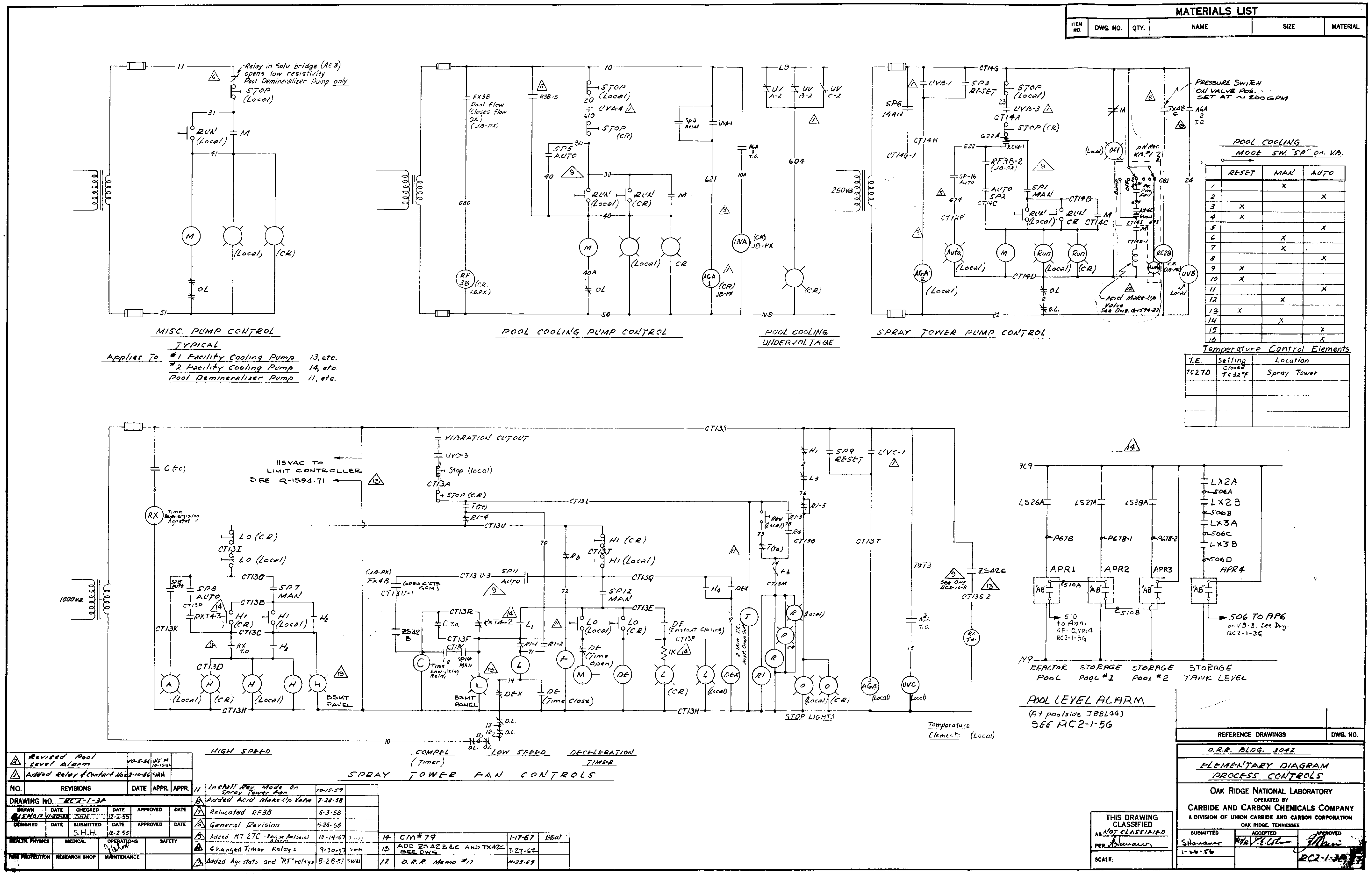
O.R.R. BLDG. 3042

ELEMENTARY DIAGRAM PROCESS CONTROLS

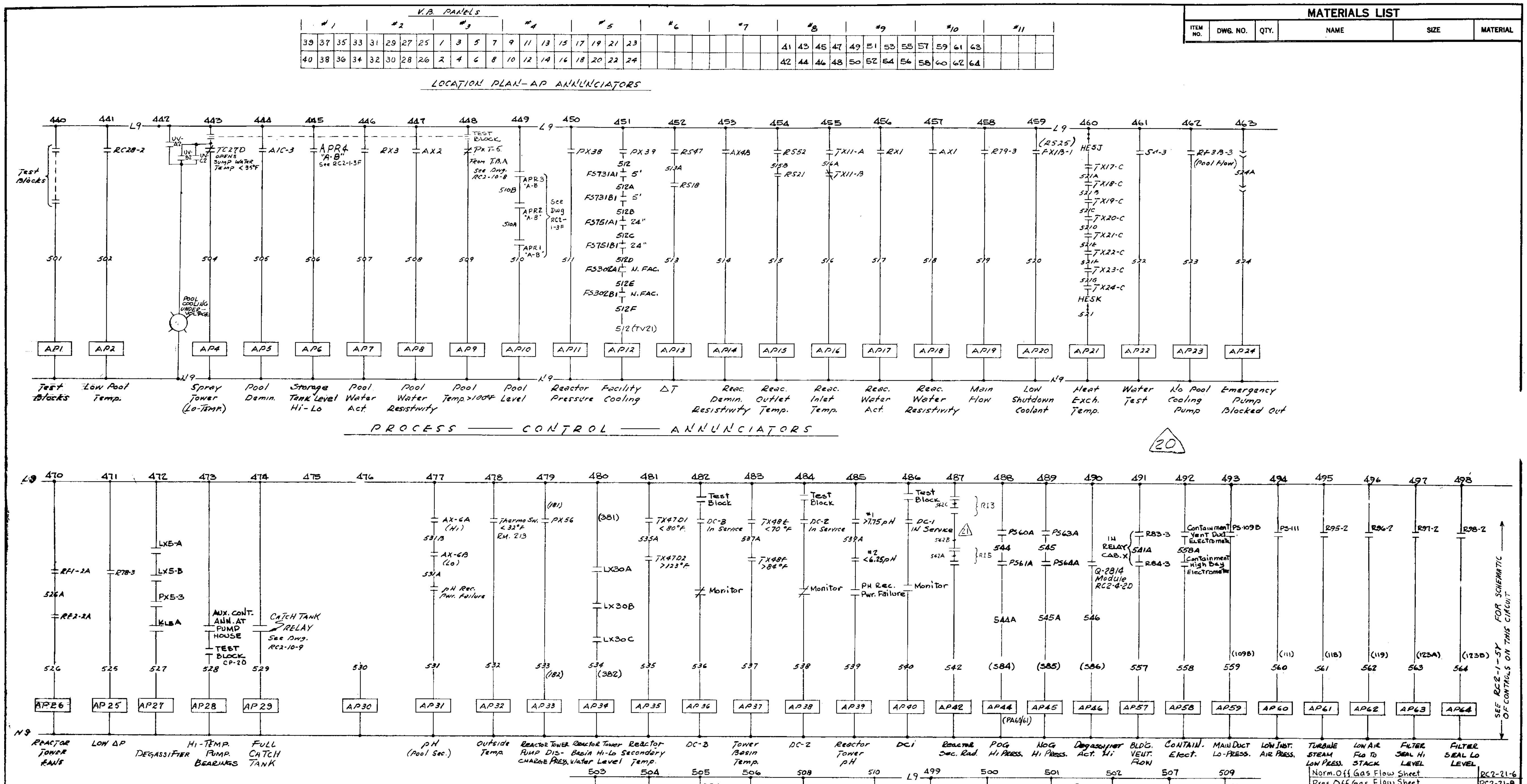
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
CARBIDE AND CARBON CHEMICALS COMPANY
A DIVISION OF UNION CARBIDE AND CARBON CORPORATION
OAK RIDGE, TENNESSEE

SUBMITTED	ACCEPTED	APPROVED
S. Hanauer 1-26-56		

RC2-1-3E REV. 9



MATERIALS LIST
ITEM NO. | DWG. NO. | QTY. | NAME | SIZE | MATERIAL
MISC. PUMP CONTROL
TYPICAL
Applies To #1 Facility Cooling Pump 13, etc.
#2 Facility Cooling Pump 14, etc.
Pool Demineralizer Pump 11, etc.
Relay in Solu bridge (AE3) opens low resistivity Pool Demineralizer Pump only
POOL COOLING PUMP CONTROL
POOL COOLING UNDERVOLTAGE
SPRAY TOWER PUMP CONTROL
PRESSURE SWITCH ON VALVE POS. SET AT ~ 200 GPM
Acid Make-Up Valve See Dwg. Q-1594-37
POOL COOLING MODE SW. "SP" on. VB.
RESET | MAN | AUTO
Temperature Control Elements
T.E. | Setting | Location
TC27D | Closed T<32°F | Spray Tower
115VAC TO LIMIT CONTROLLER SEE Q-1594-71
VIBRATION CUTOUT
HIGH SPEED
COMPEL (Timer)
LOW SPEED
DECELERATION TIMER
SPRAY TOWER FAN CONTROLS
STOP LIGHTS
Temperature Elements (Local)
REACTOR Pool
STORAGE POOL #1
STORAGE Pool #2
STORAGE TANK LEVEL
POOL LEVEL ALARM
(At poolside JBBL44)
SEE RC2-1-56
Revised Pool Level Alarm 10-5-56
Added Relay & Contact No. 22 12-10-56 SHH
Install Rev. Mode on Spray Tower Fan 10-15-59
Added Acid Make-Up Valve 7-28-58
Relocated RF3B 6-3-58
General Revision 5-26-58
Added RT 27C 10-14-57
Changed Timer Relay 9-30-57
Added Agastats and "RT" relays 8-28-57 SWM
14 CM#79 1-17-67 BBW
13 ADD ZS42 B&C AND TX42C SEE DWG 7-27-62
12 O.R.R. Memo #17 11-23-59
DRAWING NO. RC2-1-3A
DRAWN | CHECKED SHN 12-2-55
SUBMITTED S.H.H. 12-2-55
REFERENCE DRAWINGS | DWG. NO.
O.R.R. BLDG. 3042
ELEMENTARY DIAGRAM PROCESS CONTROLS
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
CARBIDE AND CARBON CHEMICALS COMPANY
A DIVISION OF UNION CARBIDE AND CARBON CORPORATION
OAK RIDGE, TENNESSEE
THIS DRAWING CLASSIFIED AS NOT CLASSIFIED
SCALE:
RC2-1-3A



NO.	REVISIONS	DATE	APPR.	APPR.
9	Used Annu. AP26	12-12-60		
8	30 M W REVISIONS	7-5-60		
7	O.R.R. Memo #17 & 21	11-28-59		
6	Used Annun AP2 & AP23	5-14-58		
5	Used Annun AP1	10-5-56		
4	54-3 Was 54-7	2-10-56	SHH	

NO.	REVISIONS	DATE	APPR.
21	CM-85 Change R1M 3 & 5 Alarm ckt. to fit new mon.	22 Jan 68	
20	CHANGE MEMO ORR No. 84	7-19-67	CEM
19	CHANGE MEMO ORR NO. 82	6-14-67	CEM
18	ADDED WIND SPEED ALARM (AP-55) "AS BUILT"	6-12-67	CEM
17	C. M. ORR No. 80	3-5-67	C.E.M.
16	C.M. # 79	1-3-67	DAW
15	C.M. #75 CONTAINMENT		
14	L9 REV. CM#74	6-12-65	DAW
13	AS BUILT	6-7-65	DOB
12	ADDED NEW FILTER PIT ALARMS	6-15-64	CEM
11	Added Charcoal Filter Pit Temp Alarm ORR Memo # 58	5-3-63	
10	REMOVED SCRUBBER ELECTROMETER FROM AP30 AND LOW BLD'G. VENT FROM AP41 – MOVED TO VB10; ADD SCRUBBER ALARMS.	7-27-62	

DRAWING NO. RC2-1-3G

DRAWN	DATE	CHECKED	DATE	APPROVED	DATE
BISHOP	12-2-55	SHH	12-2-55		
DESIGNED	**DATE**	**SUBMITTED**	**DATE**	**APPROVED**	**DATE**
		S.H.H.	12-2-55		

HEALTH PHYSICS | MEDICAL | OPERATIONS | SAFETY
FIRE PROTECTION | RESEARCH SHOP | MAINTENANCE

REFERENCE DRAWINGS	DWG. NO.
Norm. Off Gas Flow Sheet	RC2-21-6
Pres. Off Gas Flow Sheet	RC2-21-8
Process Flow Sheets	RC2-21-1(-)
CELL VENT. ELEMENTARY & WIRING	RC2-1-5Y

THIS DRAWING CLASSIFIED AS NOT CLASSIFIED PER Hanauer

SCALE:

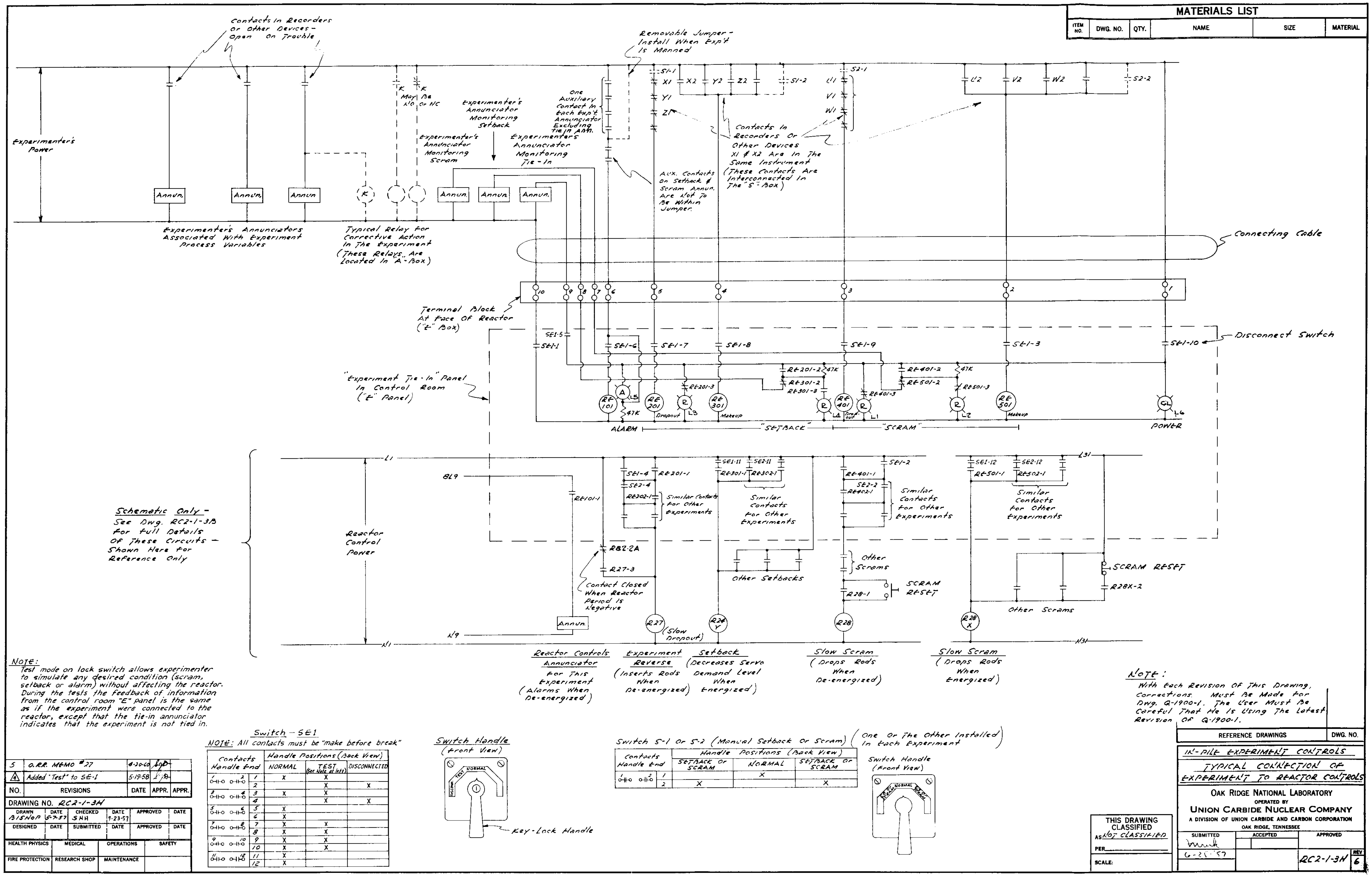
O.R.R. BLDG. 3042

ELEMENTARY DIAGRAM
PROCESS ANNUNCIATORS

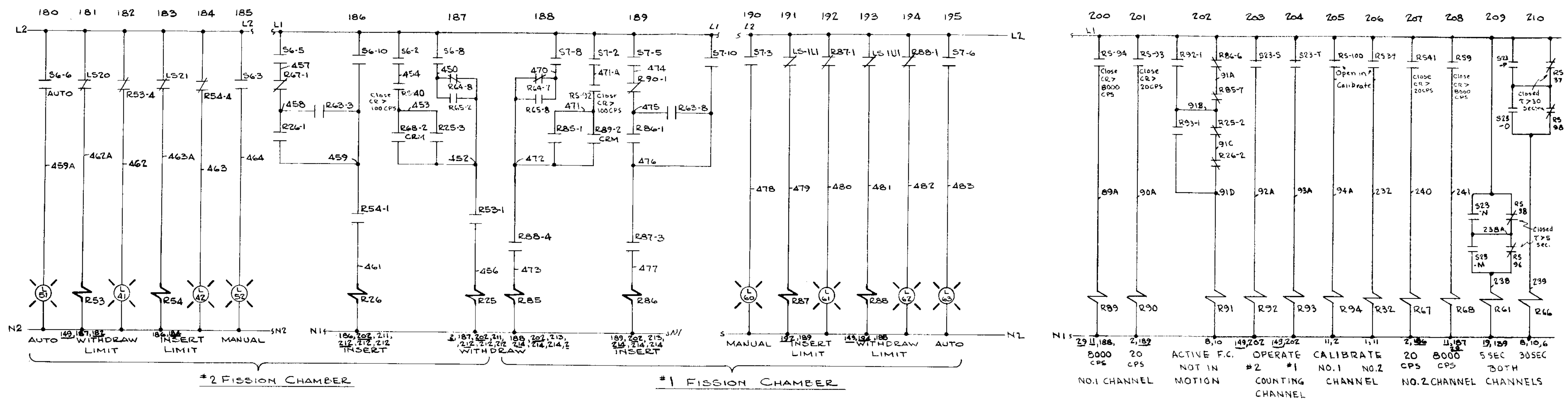
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
CARBIDE AND CARBON CHEMICALS COMPANY
A DIVISION OF UNION CARBIDE AND CARBON CORPORATION
OAK RIDGE, TENNESSEE

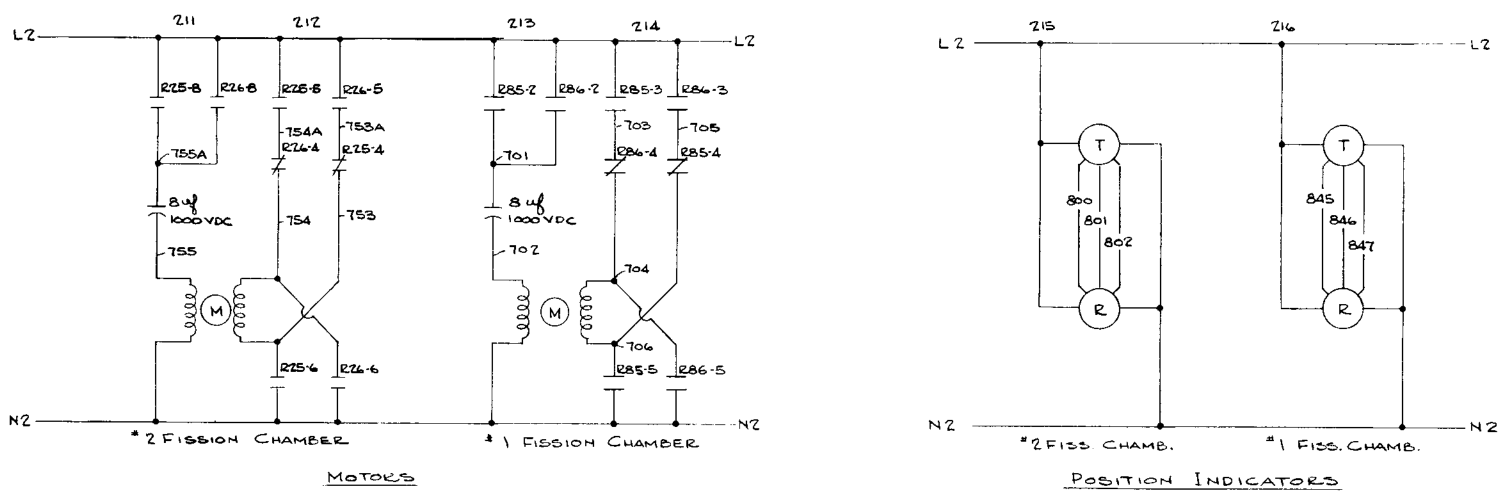
SUBMITTED	ACCEPTED	APPROVED
S. Hanauer 1-26-56	1/27/56 T. E. Cole	

RC2-1-3G 21



MATERIALS LIST
ITEM NO. | DWG. NO. | QTY. | NAME | SIZE | MATERIAL
Contacts In Recorders Or Other Devices - Open On Trouble
Experimenter's Power
Annun.
Experimenter's Annunciators Associated With Experiment Process Variables
Typical Relay For Corrective Action In The Experiment (These Relays Are Located In "A"-Box)
K May Be NO or NC
Experimenter's Annunciator Monitoring Setback
Experimenter's Annunciator Monitoring Scram
Experimenter's Annunciator Monitoring Tie-In
One Auxiliary Contact In Each Exp't Annunciator Excluding Tie-In Ann.
Removable Jumper - Install When Exp't Is Manned
Aux. Contacts On Setback & Scram Annun. Are Not To Be Within Jumper.
Contacts In Recorders Or Other Devices X1 & X2 Are In The Same Instrument (These Contacts Are Interconnected In The "S"-Box)
Connecting Cable
Terminal Block At Face Of Reactor ("E" Box)
Disconnect Switch
"Experiment Tie-In" Panel In Control Room ("E" Panel)
ALARM
"SETBACK"
"SCRAM"
POWER
Dropout
Makeup
Schematic Only - See Dwg. RC2-1-3B For Full Details Of These Circuits - Shown Here For Reference Only
Reactor Control Power
Similar Contacts For Other Experiments
Other Setbacks
Other Scrams
SCRAM RESET
Contact Closed When Reactor Period Is Negative
Slow Dropout
Reactor Controls Annunciator For This Experiment (Alarms When De-energized)
Experiment Reverse (Inserts Rods When De-energized)
Setback (Decreases Servo Demand Level When Energized)
Slow Scram (Drops Rods When De-energized)
Slow Scram (Drops Rods When Energized)
NOTE: Test mode on lock switch allows experimenter to simulate any desired condition (scram, setback or alarm) without affecting the reactor. During the tests the feedback of information from the control room "E" panel is the same as if the experiment were connected to the reactor, except that the tie-in annunciator indicates that the experiment is not tied in.
NOTE: With Each Revision Of This Drawing, Corrections Must Be Made For Dwg. Q-1900-1. The User Must Be Careful That He Is Using The Latest Revision Of Q-1900-1.
Switch - SE1
NOTE: All contacts must be "make before break"
Contacts Handle End | Handle Positions (Back View): NORMAL | TEST (See Note at left) | DISCONNECTED
Switch Handle (Front View)
Key-Lock Handle
Switch S-1 Or S-2 (Manual Setback Or Scram) (One Or The Other Installed In Each Experiment)
Contacts Handle End | Handle Positions (Back View): SETBACK OR SCRAM | NORMAL | SETBACK OR SCRAM
5 | O.R.R. MEMO #27 | 4-20-60
Added "Test" to SE-1 | 5-19-58
NO. | REVISIONS | DATE | APPR. | APPR.
DRAWING NO. RC2-1-3H
DRAWN BISHOP 5-7-57 | CHECKED SHH | DATE 9-23-57 | APPROVED | DATE
DESIGNED | DATE | SUBMITTED | DATE | APPROVED | DATE
HEALTH PHYSICS | MEDICAL | OPERATIONS | SAFETY
FIRE PROTECTION | RESEARCH SHOP | MAINTENANCE
THIS DRAWING CLASSIFIED AS NOT CLASSIFIED
PER
SCALE:
REFERENCE DRAWINGS | DWG. NO.
IN-PILE EXPERIMENT CONTROLS
TYPICAL CONNECTION OF EXPERIMENT TO REACTOR CONTROLS
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY
A DIVISION OF UNION CARBIDE AND CARBON CORPORATION
OAK RIDGE, TENNESSEE
SUBMITTED | ACCEPTED | APPROVED
6-25-57
RC2-1-3H | REV 6





RS #	INSTRUMENT	CKT NO.
9	#2 Log C.R. Rcrdr	208
37	#2 Count Rate T Rcrdr	210
38	#2 " " " "	209
39	#2 Log CR Ampl.	206
40	#2 Log CR Rcrdr	187
41	#2 " " "	207
92	#1 Log C.R. Rcrdr	188
93	#1 Log CR Rcrdr	201
94	#1 " " "	200
96	#1 Count Rate T Rcrdr	209
98	#1 " " " "	210
100	#1 Log CR Ampl.	205

REFERENCE DRAWINGS	DWG. NO.
RELAY PANEL WIRING	RC2-5-2C
CONTROL ELEM.	RC2-1-3A&B

DESIGN RESPONSIBILITY A.E.G. Bates

O.R.R. BLD'G. 3042

ELEMENTARY DIAGRAM
FISSION CHAMBER DRIVE
CKTS. 180-220

INSTRUMENTATION AND CONTROLS DIVISION
OAK RIDGE NATIONAL LABORATORY
OPERATED BY
UNION CARBIDE NUCLEAR COMPANY

LIMITS ON DIMENSIONS UNLESS OTHERWISE SPECIFIED
FRACTIONS: ±
DECIMALS: ±
ANGLES: ±
SCALE:

SUBMITTED | ACCEPTED | DATE | APPROVED | DATE

RC2-1-31 REV 1

NO.	RC2-1-31 REVISIONS	DATE	APPD	APPD
1	Add Ckt Nos. & Ref info	6 June 63	AEGB	6-18-63

DRAWN	DATE	SUBMITTED	DATE	APPROVED	DATE
D.a. Baker	9-18-62			A.E.G.B	25 Feb 63
DESIGNED	DATE	APPROVED	DATE	APPROVED	DATE
CHECKED	DATE	APPROVED	DATE	APPROVED	DATE

RELAY NO.	25	26	53	54	63	64	65	67	68	85	86	87	88	89	90	92	93
CIRCUIT NO.	187	186	181	183	1	2	60	207	208	188	189	191	193	200	201	203	204
RELAY NO.	32	61	66	94													
CIRCUIT NO.	206	209	210	205													

LS NO.	20	21	1L1	101
CIRCUIT NO.	181	183	191	193

DIETZGEN NO. 196-H ALBEPROOF

Appendix E

BIBLIOGRAPHY OF ORR REPORTS[1]

This bibliography contains references and listings of published reports and articles related to the ORR. It is not to be assumed that the bibliography is complete, but it should serve as a guide in obtaining information about the reactor system, either as it was designed and built or as it is now. In using this bibliography and the references listed herein, one should keep in mind that, after the reactor system was designed and even after its construction, extensive modifications have been made to the system. Hence, some of the earliest reports and descriptions of components are no longer strictly valid and may be somewhat misleading.

In order to present the references in a readily usable form, they have been grouped into subject categories as follows:

1. Auxiliaries
2. Construction and Building Features; General Descriptions
3. Control and Safety System and Components
4. Cooling System
5. Criticality, Reactivity, and Core Calculations
6. Emergency Systems
7. Experimental Facilities
8. Fuel Elements
9. Gamma Heating
10. Heat Transfer
11. Instrumentation
12. Isotope Production
13. Neutron Spectrum and Distribution
14. Operating Procedures
15. Operations and Progress Reports
16. Safety: Industrial, Radiation, and Reactor
17. Shielding and Radiation Hazards
18. Specifications
19. Stresses
20. Waste Disposal

Besides the references listed herein, the annual and quarterly reports for the ORR contain additional information about the reactor.

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[1]This bibliography was obtained, in part, from an internal publication by E. N. Cramer and K. D. George.

Report Number	Date	Title	Author
		1. Auxiliaries	
CF-55-8-183	August 18, 1955, 78 pp.	*Auxiliary Pumps for the ORR Process Water System*	J. P. Sanders and H. L. Watts
CF-59-8-50	August 12, 1959, 20 pp.	*Secondary-Side Water Treatment for Corrosion Control in Al Heat Exchangers*	P. D. Neumann
		2. Construction and Building Features; General Descriptions	
CF-51-8-216	August 30, 1951, 33 pp.	*ORNL Research Reactor Design Data*	W. E. Sholl, Jr.
CF-54-3-49	March 17, 1954, 35 pp.	*ORNL Research Reactor (ORR): Preliminary Proposal No. 179*	
CF-54-5-27	May 5, 1954, 8 pp.	*Effect of Pool Temperature on Building Humidity for the ORR*	J. P. Sanders
CF-56-5-186	April 19, 1956, 58 pp.	*The ORR Reactor Building, Reactor Structure and Services*	W. L. Wright
ORNL-1475	January 22, 1953, 51 pp.	*ORNL Research Reactor Project (ORR)*	J. P. Gill, comp.
ORNL-2240	February 8, 1957, 63 pp.	*The Oak Ridge National Laboratory Research Reactor (ORR): A General Description*	T. E. Cole and J. P. Gill
Nuclear Engineering	*Chem. Eng. Progr., Symp. Ser.* **12**, Pt. 2, 43–58 (1954), 15 pp.	*A High Performance Research Reactor*	J. P. Gill
Nucleonics	*Nucleonics* **16**(8), 120A (1958), 1 p.	*Reactors on the Line, ORR*	
Peaceful Uses Atomic Energy	*Proc. U.N. Intern. Conf. Peaceful Uses At. Energy, 2nd, Geneva, 1958*, **10**, 86, 46 pp.	*Design and Operation of the ORR*	T. E. Cole and J. A. Cox
Nuclear Science and Engineering	*Nucl. Sci. Eng.* **2**(1), Suppl. 72–3 (1959), 2 pp.	*The Oak Ridge Research Reactor: III. Normal Operation*	J. A. Cox *et al.*
ORNL-2086	August 2, 1956, 11 pp.	*A Method for the Disposal of Volatile Fission Products from an Accident in the Oak Ridge Research Reactor*	F. T. Binford and T. H. J. Burnett

Report Number	Date	Title	Author
ORNL-1794	October 7, 1954	*The Oak Ridge National Laboratory Research Reactor. Safeguard Report, Vol I and Vol II — Appendix*	F. T. Binford

3. Control and Safety System and Components

Report Number	Date	Title	Author
CF-53-11-25	December 8, 1953, 7 pp.	*Release Time Investigation of the Oak Ridge Research Reactor Shim Rod Support*	L. C. Oakes
CF-57-9-48	September 6, 1957, 99 pp.	*Design and Experimental Evaluation of Electro-magnets for Research Reactors*	C. Michelson
CF-58-2-3	February 14, 1958, 47 pp.	*Development of a Fast-Release Electromagnet for Pool-Type Research Reactor*	C. Michelson
ORNL-2898	April 1, 1960, 9 pp.	*Photoneutrons and the Control of a Pool-Type Reactor*	A. L. Colomb
CF-60-11-75	November 4, 1960, 13 pp.	*Experimental Determination of an Adequate Fission Chamber Location in the ORR Pool*	D. P. Roux and A. L. Colomb
ORNL-TM-283	August 28, 1962	*Reactor Controls Reliability and Maintenance at the ORR*	K. W. West
ORNL-TM-605	August 6, 1963	*Experience with the Use of the Rod-Drop Method of Rod Calibration at the ORR and LITR*	F. R. Buoni

4. Cooling System

Report Number	Date	Title	Author
CF-52-11-148	November 20, 1952, 18 pp.	*ORNL Research Reactor Cooling System Cost Estimates for 5, 15, and 30 Megawatt Power Levels*	F. C. McCullough
CF-53-3-204	March 25, 1953, 4 pp.	*Cooling Requirements for the ORR*	R. B. Briggs
CF-54-3-169	March 19, 1954, 11 pp.	*Operating Conditions of the ORR Cooling System*	J. P. Sanders
CF-55-3-84	March 15, 1955, 9 pp.	*Large Diameter Piping in the Basement of the ORR Building*	J. P. Sanders

Report Number	Date	Title	Author
CF-55-5-125	May 19, 1955, 19 pp.	*A Functional Outline of the Process Water System for the ORR*	J. P. Sanders and H. L. Watts
CF-55-5-142	May 23, 1955, 12 pp.	*Operating Conditions of the ORR Cooling System at 20 Mw*	J. P. Sanders
CF-55-7-122	July 8, 1955, 8 pp.	*Pumps for Main Circulating Loop of the ORR*	J. P. Sanders and H. L. Watts
CF-60-4-97	April 25, 1960, 18 pp.	*Corrosion in the Oak Ridge Research Reactor Core-Cooling System*	P. D. Neumann
CF-59-8-50	August 12, 1959, 20 pp.	*Secondary-Side Water Treatment for Corrosion Control in Al Heat Exchangers*	P. D. Neumann
CF-59-8-59	August 14, 1959, 20 pp.	*Shutdown Cooling of ORR*	T. E. Cole
CF-59-9-68	1959, 12 pp.	*Notes on Heat Transfer in the ORR Core at Powers Greater than 20 Mw*	J. F. Wett, Jr.
CF-60-2-61	February 15, 1960, 23 pp.	*Natural Circulation Burnout Heat Flux for the ORR*	J. F. Wett, Jr.
CF-60-6-13	June 7, 1960, 6 pp.	*Requirements for Afterheat Removal for 30 Mw Operation of the ORR*	J. F. Wett, Jr.
ORNL-2311	July 2, 1957, 172 pp.	*Radionuclides in Reactor Cooling Water — Identification, Source, and Control*	D. W. Moeller
CF-58-2-11	February 17, 1958	*Preliminary Report on the Results of the Oak Ridge Research Reactor Hydraulic Test*	F. T. Binford
CF-54-4-239		*Water Borne Activity in the Cooling System of the ORNL Research Reactor*	F. T. Binford
CF-55-5-157		*Hydraulic and Thermal Characteristics of the ORR Core*	F. T. Binford

5. Criticality, Reactivity, and Core Calculations

Report Number	Date	Title	Author
CF-55-8-175	August 31, 1955, 17 pp.	*ORR Reactor Physics: Problem I, Infinite Slab Model*	M. L. Nelson and N. F. Lansing
CF-57-5-31	March 11, 1958, 20 pp.	*Two Group Calculations for Flux Distribution and Critical Mass in Clean Cold ORR Cores*	F. T. Binford

Report Number	Date	Title	Author
CF-58-9-40	October 2, 1958, 6 pp.	*Critical Experiments with Arrays of ORR and BSR Fuel Elements*	J. K. Fox and L. W. Gilley
ORNL-1682	November 19, 1954, 79 pp.	*Reactivity Measurements with the Bulk Shielding Reactor*	R. G. Cochran, J. L. Meem, T. E. Cole, and E. B. Johnson
ORNL-1871	September 6, 1955, 34 pp.	*Reactivity Measurements with the Bulk Shielding Reactor: Control Rod Calibrations, Beam Hole Coefficients, Partial Reflector Coefficients*	E. B. Johnson, F. C. Maienschein, K. M. Henry, R. G. Cochran, and J. D. Flynn
ORNL-2559	June 10, 1959, 56 pp.	*Initial Post-Neutron Measurements in the ORR*	C. D. Cagle and R. A. Costner, Jr.
ORNL-2897	May 19, 1960, 9 pp.	*Fission Product Distribution in ORR Fuel Elements*	A. L. Colomb
ORNL-2898	April 1, 1960, 9 pp.	*Photoneutrons and the Control of a Pool-Type Reactor*	A. L. Colomb
ORNL-TM-277	July 31, 1962	*Fuel Cycles and Loading Programming for Water-Cooled Research Reactors*	A. L. Colomb and Doyle Cavin
ORNL-TM-605	August 6, 1963	*Experience with the Use of the Rod-Drop Method of Rod Calibration at the ORR and LITR*	F. R. Buoni
CF-64-10-34	October 13, 1964	*Xenon Concentration in the ORR Core at 45-Mw Operation*	T. P. Hamrick
CF-64-10-20	October 8, 1964	*A Study of Rod Position Versus Time During Steady-State Operation of the ORR at 30 Mw*	T. P. Hamrick
CF-64-10-28	October 19, 1964	*Comparison of New and Used ORR Fuel Elements*	T. P. Hamrick
CF-64-10-19	October 1, 1964	*ORR Fuel Units – Material Composition, Cross Sections, and Calculation of* k_∞	T. P. Hamrick

Report Number	Date	Title	Author
		An Experimental Evaluation of Errors Arising from the Burnup Calculation Used at the ORNL Research Reactor, American Nuclear Society Conference on Reactor Operating Experience, Jackson, Wyoming, July 28–29, 1965	T. P. Hamrick
		Experience in Computing Research Reactor Fuel Burnup Contrasted with Recovery Results, presented at 1966 AEC contractor Nuclear Materials Management Meeting, October 3–5, 1966	T. P. Hamrick and H. F. Stringfield

6. Emergency Systems

Report Number	Date	Title	Author
ORNL-2086	August 2, 1956, 11 pp.	*A Method for the Disposal of Volatile Fission Products from an Accident in the Oak Ridge Research Reactor*	F. T. Binford and T. H. J. Burnett
CF-58-5-59	May 21, 1958, 13 pp.	*Proposed Method for Removal of Radio-Iodine Vapor from Experiment Off-Gas System of the ORR*	R. E. Adams and W. E. Browning, Jr.
ORNL-TM-273	August 23, 1962	*Vapor Containment in the Oak Ridge Research Reactor*	F. T. Binford

7. Experimental Facilities

Report Number	Date	Title	Author
CF-53-4-48	April 8, 1953, 7 pp.	*The ORR as an Engineering Test Facility*	A. M. Weinberg
CF-54-3-196	March 31, 1954, 35 pp.	*Proposed In-Pile Loop Experiment*	G. H. Jenks, D. T. Jones, J. N. Baird, and J. L. Redford
CF-56-8-169	August 27, 1956, 21 pp.	*Neutron and Gamma Flux Problems Associated with the CTD-HRP In-Pile Loop*	E. D. Arnold
CF-56-10-104	October 23, 1956, 18 pp.	*Preliminary Report on Instrumentation for ORR HRP Loop HN-1*	R. A. Lorenz

Report Number	Date	Title	Author
CF-56-12-65	December 18, 1956, 18 pp.	*Laboratory Test of a Core Cooler for ORR Loop HN-1*	R. A. Lorenz and D. T. Jones
CF-58-7-48	July 7, 1958, 53 pp.	*Analog Computer Study of the Operation of the Gas-Cooled Loop on the Oak Ridge Research Reactor*	F. P. Green, F. H. Neill, B. E. Short, and M. L. Winton
CF-58-10-21	October 23, 1958, 41 pp.	*Preliminary Design Report for the NMSR Pressurized Water Loop at the ORR*	J. T. Dudley, D. B. Trauger, and H. W. Savage
CF-58-10-59	October 20, 1958, 9 pp.	*Heating and Cooling Tests on the ORR HN-1 In-Pile Loop Mockup*	A. J. Shor
CF-58-12-10	December 23, 1958, 18 pp.	*Evaluation of the Iodine Vapor—Fission Gas Absorption Traps for ORR-705 Capsule Experiment: GCPR Capsule Irradiation Program*	R. E. Adams and W. E. Browning
CF-59-4-96	April 24, 1959, 13 pp.	*Estimate of Hazard Produced by Accidental Release of Gaseous Fission Products from an ORR Fused-Salt Capsule Experiment*	R. E. Adams and W. E. Browning
CF-59-7-31	July 13, 1959, 28 pp.	*Test of Heater and Cooler Concepts for GCR-ORR Loop, Design 4*	W. H. Kelley, Jr., and E. Storto
CF-59-8-57	August 17, 1959, 62 pp.	*Determination of Suitable Insulation for a 1-$^{5}/_{16}$″ Helium-Filled Annulus in the ORR Helium In-Pile Loop, Design No. 4*	R. B. Knight and R. E. Helms
CF-59-9-7	September 1, 1959, 34 pp.	*ORR Experimental Facilities*	A. R. Boynton
CF-59-10-38	October 1, 1959, 26 pp.	*Estimates of the Rate of Internal Heat Generation in Several Research Reactor Facilities*	P. H. Newell
CF-60-1-24	January 14, 1960, 21 pp.	*Evaluation of Activated Charcoal Fission Gas Absorbers Designed for the GC-ORR Loop Experiment No. 1*	R. E. Adams and W. E. Browning

Report Number	Date	Title	Author
CF-60-4-31	April 11, 1960, 12 pp.	*Studies of Improvement of Power Density in ORR Loops*	M. L. Tobias and D. R. Vondy
CF-60-11-76	November 8, 1960, 24 pp.	*Analog Study of the Reference Design of the Gas-Cooled ORR Loop No. 1*	S. J. Ball
CF-61-4-38	April 14, 1961, 8 pp.	*Re-Evaluation of the Activated Charcoal Fission-Gas Absorber for the Leak-Detection System of the GC-ORR Loop No. 1*	R. E. Adams and W. E. Browning
CF-61-6-3	June 1, 1961	*Volume I, EGCR Experimental Loops, Design Criteria*	
CF-61-3-11	March 3, 1961	*GC-ORR Loop II Filter Tests*	F. A. Flint and R. E. MacPherson
CF-61-3-104	March 1, 1961	*Pneumatic Tube System — Oak Ridge Research Reactor*	G. A. Cristy
CF-61-6-4	December 29, 1961	*Volume I, EGCR Experimental Loops. Design Analyses*	
CF-61-8-67	August 23, 1961, 19 pp.	*GC-ORR Loop No. 2 Check Valve Tests*	F. A. Flint
CF-61-12-16	December 5, 1961	*In-Pile Slurry Loop Facility O-1-28S in ORR Beam Hole HN-1: Request for Approval*	A. J. Shor, R. A. Lorenz, *et al.*
ORNL-1965	January 22, 1957, 56 pp.	*A Fluoride Fuel In-Pile Loop Experiment*	O. Sisman, W. E. Brundage, and W. W. Parkinson
ORNL-3050	February 9, 1961, 28 pp.	*Experiment on Continuous Release of Fission Gas During Radiation*	R. M. Carroll and C. D. Baumann
ORNL-3096	September 15, 1961, 37 pp.	*Measurements at Beam Hole HB-2 of the Oak Ridge Research Reactor for South Facility Movable Shield Design*	F. J. Muckenthaler, T. V. Blosser, J. M. Miller, and L. Jung
AECU-3819	March 1958, 23 pp.	*Reactivity Effect of a PPR Loop in ORR*	P. F. Palmedo, E. C. Hansen, and C. D. Koch
Symp. Franklin Inst., Philadelphia, 1960	*J. Franklin Inst., Monograph Ser.* No. 7, 277–92, 15 pp.	*Design of Experimental Gas-Cooled In-Pile Loops*	E. Storto
CF-59-4-70	April 17, 1959, 25 pp.	*HRP In-Pile Solution Loop Facility in ORR Beam Hole HN-1: Request for Approval*	R. A. Lorenz, H. C. Savage, and J. R. McWherter

Report Number	Date	Title	Author
CF-59-7-93	March 4, 1960, 24 pp.	*Guide for the Design of Safety Instrumentation for Experiments in the LITR and ORR*	J. T. Lorenzo
CF-57-12-58	December 17, 1957, 17 pp.	*Safety Procedures for In-Pile Experiments in the LITR and the ORR*	C. Brashear and S. H. Hanauer
	October 1963	*Design and Safety Analysis of Experiments in ORNL Research Reactors,* presented at Conference on the Problems of Operating Research and Power Reactors (ANS), Ottawa, Canada	R. A. Costner, Jr., and L. E. Stanford
ORNL-TM-266	August 17, 1962	*Experiment Facilities of the Oak Ridge Graphite Reactor*	S. D. Sheppard
ORNL-TM-279	August 28, 1962	*The Oak Ridge Research Reactor (ORR), The Low-Intensity Testing Reactor (LITR), and the ORNL Graphite Reactor (OGR) as Experiment Facilities*	K. D. George
ORNL-TM-399		*A Facility of High Thermal Neutron Flux in the Absence of Fast Neutrons and Gamma Rays*	R. van der Walt and A. L. Colomb
CF-58-12-158	December 1958	*Safety and Operability Review of Experiments to Be Operated in Nuclear Reactors at ORNL*	C. D. Cagle
ORNL-TM-281	August 20, 1962	*General Standards Guide for Experiments in ORNL Research Reactors*	C. D. Cagle
ORNL-TM-745	1964	*Considerations Involved in the Safety Review of Experiments to Be Operated in Nuclear Reactors*	C. D. Cagle

8. Fuel Elements

Report Number	Date	Title	Author
CF-57-6-87	July 15, 1957, 22 pp.	*The Volatilization of Fission Products by Melting of Reactor Fuel Plates*	G. W. Parker and G. E. Creek
CF-59-8-103	August 26, 1959, 10 pp.	*A Preliminary Report on Irradiated ORR Fuel-Element Temperatures in Air*	J. F. Wett, Jr.

Report Number	Date	Title	Author
ORNL-2616	July 7, 1959, 28 pp.	*Experiments on the Release of Fission Products from Molten Reactor Fuels*	G. E. Creek, W. J. Martin, and G. W. Parker
ORNL-2892	April 6, 1960, 15 pp.	*Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air*	J. F. Wett, Jr.
ORNL-2939	April 29, 1960, 46 pp.	*Effect of Heat Flux on the Corrosion of Aluminum by Water, Part 1. Experimental Equipment and Preliminary Test Results*	J. C. Griess, H. C. Savage, T. H. Mauney, and J. L. English
ORNL-3151	June 23, 1961	*The Corrosion of Aluminum Alloys in the Oak Ridge Research Reactor*	P. D. Neumann
ORNL-2897	May 19, 1960, 9 pp.	*Fission Product Distribution in ORR Fuel Elements*	A. L. Colomb
Nuclear Science and Engineering	*Nucl. Sci. Eng.* 9, 96–98 (January 1960)	*Local Flux Distributions in ORR Fuel Elements*	A. L. Colomb and J. F. Wett, Jr.
CF-64-10-28	October 19, 1964	*Comparison of New and Used ORR Fuel Elements*	T. P. Hamrick
		An Experimental Evaluation of Errors Arising from the Burnup Calculation Used at the ORNL Research Reactor, American Nuclear Society Conference on Reactor Operating Experience, Jackson, Wyoming, July 28–29, 1965	T. P. Hamrick
CF-64-10-19	October 19, 1964	*ORR Fuel Units – Material Composition, Cross Sections and Calculation of* k_∞	T. P. Hamrick
		Experience in Computing Research Reactor Fuel Burnup Contrasted with Recovery Results, presented at 1966 AEC contractor Nuclear Materials Management Meeting, October 3–5, 1966	T. P. Hamrick and H. F. Stringfield
	Nucl. Safety 5(2) (1964)	*ORR Fuel Failure Incident*	W. H. Tabor and T. M. Sims

Report Number	Date	Title	Author
		Fuel Plate Melting at the ORR, American Nuclear Society Conference on Reactor Operating Experience, Jackson, Wyoming, July 28–29, 1965	W. H. Tabor
ORNL-TM-627	October 1964	*Report on Fuel-Plate Melting at the Oak Ridge Research Reactor, July 1, 1963*	W. H. Tabor and T. M. Sims
CF-6-10-80		*Purchase of ORR Fuel Assemblies*	F. T. Binford

9. Gamma Heating

Report Number	Date	Title	Author
CF-55-7-134	July 29, 1955, 20 pp.	*Energy Absorption Rate and Temperature Distributions in the Reactor Tank and Experimental Facilities*	J. P. Sanders
CF-55-10-19	October 14, 1955, 18 pp.	*Temperature Distribution in the ORR Core Housing*	F. T. Binford
CF-56-3-31	March 7, 1956, 5 pp.	*Surface Temperatures in the ORR Tank near the Poolside Facility*	E. S. Bettis and F. T. Binford
CF-56-3-72	March 7, 1956, 20 pp.	*Gamma Heating Measurements in the Bulk Shielding Reactor*	F. T. Binford, E. S. Bettis, and J. T. Howe
CF-59-4-70	April 17, 1959, 25 pp.	*HRP In-Pile Solution Loop Facility in ORR Beam Hole HN-1: Request for Approval*	R. A. Lorenz, H. C. Savage, and J. R. McWherter
CF-56-8-169	August 27, 1956, 21 pp.	*Neutron and Gamma Flux Problems Associated with the CTD-HRP In-Pile Loop*	E. D. Arnold

10. Heat Transfer

Report Number	Date	Title	Author
CF-54-3-57	March 11, 1954, 36 pp.	*Heat Transfer and Pressure Loss in Proposed ORR Fuel Assemblies*	J. P. Sanders
CF-54-4-170	April 27, 1954, 34 pp.	*Free Convection in the Reactor Pool (Revised)*	N. F. Lansing
CF-55-5-157	May 27, 1955, 8 pp.	*Hydraulic and Thermal Characteristics of the ORR Core*	F. T. Binford
CF-55-8-42	August 8, 1955, 16 pp.	*Flow in Cooling Channels of the Large Facility and Beam Hole Liners and Core Housing*	J. P. Sanders
CF-56-1-10	January 3, 1956, 23 pp.	*Heat Removal from Ionization Chamber Shield*	N. F. Lansing

Report Number	Date	Title	Author
CF-56-4-27	April 5, 1956, 8 pp.	*Hydraulic Test of ORR Dummy Fuel Element*	E. S. Bettis
CF-58-2-11	February 17, 1958, 8 pp.	*Preliminary Report on the Results of the Oak Ridge Research Reactor Hydraulic Test*	F. T. Binford
CF-59-8059	August 14, 1959, 20 pp.	*Shutdown Cooling of ORR*	T. E. Cole
CF-59-9-68	1959, 12 pp.	*Notes on Heat Transfer in the ORR Core at Powers Greater than 20 Mw*	J. F. Wett, Jr.
CF-60-2-61	February 15, 1960, 23 pp.	*Natural Circulation Burnout Heat Flux for the ORR*	J. F. Wett, Jr.
CF-60-6-13	June 7, 1960, 6 pp.	*Requirements for Afterheat Removal for 30-Mw Operation of the ORR*	J. F. Wett, Jr.
CF-61-1-6	January 4, 1961, 18 pp.	*ORR Startup Accident and Cooling Flow Coastdown Analog Analysis*	R. S. Stone and A. L. Colomb
CF-61-1-49	January 19, 1961, 18 pp.	*Some Heat Transfer Characteristics of ORR Fuel Elements*	J. F. Wett, Jr.
CF-55-10-19	October 14, 1955	*Temperature Distribution in the ORR Core Housing*	F. T. Binford
ORNL-TM-274	August 17, 1962	*The Detection of Boiling in a Water-Cooled Nuclear Reactor*	F. T. Binford and A. L. Colomb

11. Instrumentation

Report Number	Date	Title	Author
CF-59-7-93	March 4, 1960, 24 pp.	*Guide for the Design of Safety Instrumentation for Experiments in the LITR and ORR*	J. T. De Lorenzo
CF-60-11-75	November 4, 1960, 13 pp.	*Experimental Determination of an Adequate Fission Chamber Location in the ORR Pool*	D. P. Roux and A. L. Colomb
CF-56-10-104	October 23, 1956, 18 pp.	*Preliminary Report on Instrumentation for ORR HRP Loop HN-1*	R. A. Lorenz
ORNL-TM-283	August 28, 1962	*Reactor Controls Reliability and Maintenance at the ORR*	K. W. West

Report Number	Date	Title	Author
		12. Isotope Production	
		13. Neutron Spectrum and Distribution	
CF-60-11-31	November 3, 1960, 10 pp.	*Neutron Flux from the ORR Reactor Window into the Reactor Pool*	L. C. Bate, G. W. Leddicote, and F. C. Burns
ORNL-3028	February 7, 1961, 45 pp.	*Fast-Flux Measurements in the ORR Core*	P. Drogoumis, J. R. Weir, Jr., and G. W. Leddicote
Nuclear Science and Engineering	*Nucl. Sci. Eng.* 9, 96–98 (January 1960)	*Local Flux Distributions in ORR Fuel Elements*	A. L. Colomb and J. F. Wett, Jr.
CF-56-8-169	August 27, 1956, 21 pp.	*Neutron and Gamma Flux Problems Associated with the CTD-HRP In-Pile Loop*	E. D. Arnold
ORNL-TM-399		*A Facility of High Thermal Neutron Flux in the Absence of Fast Neutrons and Gamma Rays*	R. van der Walt and A. L. Colomb
ORNL-2559	May 1959	*Initial Post-Neutron Measurements in the ORR*	C. D. Cagle and R. A. Costner, Jr.
		14. Operating Procedures	
CF-60-8-46	1960, 348 pp.	*Operating Manual for the Oak Ridge Research Reactor*	S. D. Sheppard, W. H. Tabor, *et al.*
ORNL-TM-506		*Operating Manual for the Oak Ridge Research Reactor*	ORR Staff
ORNL-TM-689 (Revised)	September 1964	*Operating Safety Limits for the Oak Ridge National Laboratory Research Reactor (ORR)*	C. C. Webster
CF-62-2-14	February 8, 1962	*Status of Specific ACRS and AEC Recommendations with Respect to ORR Operations as of January 1, 1962*	F. T. Binford
		15. Operations and Progress Reports	
CF-59-8-39	September 10, 1959, 185 pp.	*ORR Operations for Period April, 1958, to April, 1959*	J. A. Cox
CF-60-9-2	September 30, 1960, 142 pp.	*ORR Operations for Period April, 1959, to April, 1960*	J. A. Cox
CF-61-4-66	April 25, 1961, 80 pp.	*The Operation of the Oak Ridge Research Reactor*	F. T. Binford, C. D. Cagle, R. A. Costner, Jr., J. A. Cox, and W. H. Tabor

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ORNL-2562	December 9, 1958, 29 pp.	*Operation of the ORNL Graphite Reactor and of the Low-Intensity Test Reactor and Preoperational Work on the ORR*	J. A. Cox and W. R. Casto
ORNL-TM-10	October 20, 1961, 105 pp.	*ORR Operations for Period April, 1960, to April, 1961*	J. A. Cox
IDO-16520	Pp. 37–75, 1959, 39 pp.	*Operation of the Oak Ridge Research Reactor*	W. H. Tabor *et al.*
ORNL-TM-276	August 20, 1962	*Administration of ORNL Research Reactors*	W. R. Casto
ORNL-TM-275	August 23, 1962	*Problems Encountered During Four Years of ORR Operation*	R. A. Costner, Jr., and W. H. Tabor
ORNL-TM-506		*Operating Manual for the Oak Ridge Research Reactor*	ORR staff
		Operating and Safety Problems in a Research Reactor, Proceedings of a Study Group Meeting, Bangkok, 12–21 December, 1962, Utilization of Research Reactors, IAEA, Vienna, 1963, Sect. III, Reactor Operational and Safety Problems, p. 101	J. A. Cox
ORNL-TM-689	September 1964	*Operating Safety Limits for the Oak Ridge National Laboratory Research Reactor (ORR)*	C. C. Webster
	Nucl. Safety 5(1), 116–23 (1963)	*Operating Experience of the Oak Ridge Research Reactor*	W. H. Tabor
IDO-16520	May 1959	*Operation of the Oak Ridge Research Reactor*	W. H. Tabor
	Trans. Am. Nucl. Soc. 2(1) (June 1959).	*The Oak Ridge Research Reactor; I. Preoperation Tests; II. Operational Problems; III. Normal Operation*	W. H. Tabor

16. Safety: Industrial, Radiation, and Reactor

Report Number	Date	Title	Author
ORNL-1794, Vol. I	October 7, 1954, 103 pp.	*The Oak Ridge National Laboratory Research Reactor. Safeguard Report, Vol. I*	F. T. Binford, ed.

Report Number	Date	Title	Author
ORNL-1794, Vol. II	October 7, 1954, 102 pp.	*The Oak Ridge National Laboratory Research Reactor. Safeguard Report, Vol. II, Appendix*	F. T. Binford, ed.
	October 1963	*Design and Safety Analysis of Experiments in ORNL Research Reactors,* presented at Conference on the Problems of Operating Research and Power Reactors (ANS), Ottawa, Canada	R. A. Costner, Jr.
CF-64-9-81	October 1, 1964	*Investigation of Radioactivity Releases to White Oak Creek*	J. F. Manneschmidt
ORNL-TM-141	October 1961	*Equipment and Procedure for Stack-Gas Monitoring at ORNL,* Seventh Air Cleaning Seminar, Oak Ridge, Tennessee	J. F. Manneschmidt
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CF-58-12-158	December 1958	*Safety and Operability Review of Experiments to Be Operated in Nuclear Reactors at ORNL*	C. D. Cagle
ORNL-TM-745	1964	*Considerations Involved in the Safety Review of Experiments to Be Operated in Nuclear Reactors*	C. D. Cagle
		Operating and Safety Problems in a Research Reactor, Proceedings of a Study Group Meeting, Bangkok, 12–21 December, 1962, Utilization of Research Reactors, IAEA, Vienna, 1963, Sect. III, Reactor Operational and Safety Problems, p. 101	J. A. Cox

Report Number	Date	Title	Author
	Nucl. Safety 5(2) (1964)	*ORR Fuel Failure Incident*	W. H. Tabor and T. M. Sims
		Fuel Plate Melting at the ORR, American Nuclear Society Conference on Reactor Operating Experience, Jackson, Wyoming, July 28–29, 1965	W. H. Tabor
	Trans. Am. Nucl. Soc. 2(1) (June 1959)	*The Oak Ridge Research Reactor; I. Preoperation Tests; II. Operational Problems; III. Normal Operation*	W. H. Tabor
ORNL-TM-627	October 1964	*Report on Fuel-Plate Melting at the Oak Ridge Research Reactor, July 1, 1963*	W. H. Tabor and T. M. Sims
ORNL-TM-278	1962	*Personnel Exposure and Contamination Control in the Routine Operation of ORNL Research Reactors*	T. M. Sims
		17. Shielding and Radiation Hazards	
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CF-52-4-157	April 28, 1952, 7 pp.	*Shielding Requirements and Heat Generation in the Research Reactor*	R. Stephenson and T. L. Trent
CF-53-5-225	May 26, 1953, 28 pp.	*The Shield of the ORNL Research Reactor*	C. H. Barkelew
CF-54-4-171	April 27, 1954, 17 pp.	*An Estimate of the Contribution to the Radiation Intensity Above the Surface of the Research Reactor Due to the Presence of Nitrogen*^{16}	F. T. Binford
CF-54-7-95	August 9, 1954, 16 pp.	*ORR Shielding Studies I. Basic Design Information*	N. F. Lansing
CF-54-8-102	August 31, 1954, 11 pp.	*Oak Ridge Research Reactor Shadow Shield Experiment*	R. G. Cochran, K. M. Henry, and J. D. Flynn

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CF-58-1-102	January 28, 1958, 6 pp.	*Proposed Calculational Procedure for the ORR Reactor Beam-Hole Shielding*	S. K. Penny
CF-58-6-13	June 16, 1958, 21 pp.	*Review of the ORR Shield for 30 Mw Operation*	W. Zobel, E. G. Silver, and T. V. Blosser
ORNL-1739	July 14, 1954, 26 pp.	*Design and Placement Techniques of Barytes Concrete for Reactor Biological Shields*	E. G. Tirpak
ORNL-2311	July 2, 1957, 172 pp.	*Radionuclides in Reactor Cooling Water – Identification, Source, and Control*	D. W. Moeller
CF-64-9-81	October 1, 1964	*Investigation of Radioactivity Releases to White Oak Creek*	J. F. Manneschmidt
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		Fuel Plate Melting at the ORR, American Nuclear Society Conference on Reactor Operating Experience, Jackson, Wyoming, July 28–29, 1965	W. H. Tabor
ORNL-TM-627	October 1964	*Report on Fuel-Plate Melting at the Oak Ridge Research Reactor July 1, 1963*	W. H. Tabor and T. M. Sims
CF-54-4-239		*Water Borne Activity in the Cooling System of the ORNL Research Reactor*	F. T. Binford
ORNL-TM-278		*Personnel Exposure and Contamination in the Routine Operation of ORNL Research Reactors*	T. M. Sims

18. Specifications

Report Number	Date	Title	Author
ORNL-TM-689 (Revised)	September 1964	*Operating Safety Limits for the Oak Ridge National Laboratory Research Reactor*	C. C. Webster
CF-6-10-80		*Purchase of ORR Fuel Assemblies*	F. T. Binford
CF-62-2-14	February 8, 1962	*Status of Specific ACRS and AEC Recommendations with Respect to ORR Operations as of January 1, 1962*	F. T. Binford

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		19. Stresses	
CF-53-7-205	July 30, 1953, 9 pp.	*Calculation of Wall Stresses in the "A" Tank of Oak Ridge Research Reactor*	J. J. Wallace
CF-53-10-141	October 19, 1953, 14 pp.	*Temperature and Stress Distribution in a Sectional Cylindrical Shield as Typified by the Research Reactor*	F. T. Binford
CF-57-12-58	December 17, 1957, 17 pp.	*Safety Procedures for In-Pile Experiments in the LITR and the ORR*	C. Brashear and S. H. Hanauer
CF-60-5-109	July 22, 1960, 50 pp.	*An Investigation of the Structural Integrity of Selected Components of the Oak Ridge Research Reactor*	J. M. Corum, B. L. Greenstreet, *et al.*
CF-55-10-102		*Deflection of ORR Housing Due to Uneven Temperature Distribution*	F. T. Binford
		20. Waste Disposal	
CF-58-5-59	May 21, 1958, 13 pp.	*Proposed Method for Removal of Radio-Iodine Vapor from Experiment Off-Gas System of the ORR*	R. E. Adams and W. E. Browning, Jr.
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ORNL-TM-282	August 17, 1962	*The Disposal of Radioactive Liquid and Gaseous Waste at Oak Ridge National Laboratory*	J. F. Manneschmidt and E. J. Witkowski
ORNL-2086	August 2, 1956	*A Method for the Disposal of Volatile Fission Products from an Accident in the Oak Ridge Research Reactor*	F. T. Binford and T. H. J. Burnett
CF-54-5-229		*Criteria for Use in the Estimation of the Cost of Shipping Contaminated Reactor Fuels*	F. T. Binford

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