C. General Facility Layout
Drawing

File 2, Doc 1

ORNL-3572 (Rev. 2)

#210

238

Full Release

ornl

OAK RIDGE
NATIONAL
LABORATORY

MARTIN MARIETTA

THE HIGH-FLUX ISOTOPE REACTOR
A Functional Description
Volume 1A

[MA]NAGED BY
[MA]RTIN MARIETTA ENERGY SYSTEMS, INC.
[FOR] THE UNITED STATES
[DEP]ARTMENT OF ENERGY

ChemRisk Repository Number: 210
Document Number: ORNL 3572 Rev. 2
Title: The High-Flux Isotope Reactor; A Functional Description Volume 1A
Authors: F. T. Binford T. E. Cole E. N. Cramer
Abstract: Gives brief history of reactor and provide operational and technical capabilities of the HFR. Contains good list of references.
Reviewer: J. Buddenbaum
Document Source or Location: X-10 Lab Records
Date Document Issued: 05/00/88
Classification Category: unc
Site Document Addresses: X
Primary Document Category: HO

Date Entered: 11/13/92
Entered By: cmv
Keywords: Experiment Nuclear Shielding

ORNL-3572 (Rev. 2)
UC-80 – Reactor Technology

Contract No. W-7405-eng-26

THE HIGH-FLUX ISOTOPE REACTOR

A Functional Description

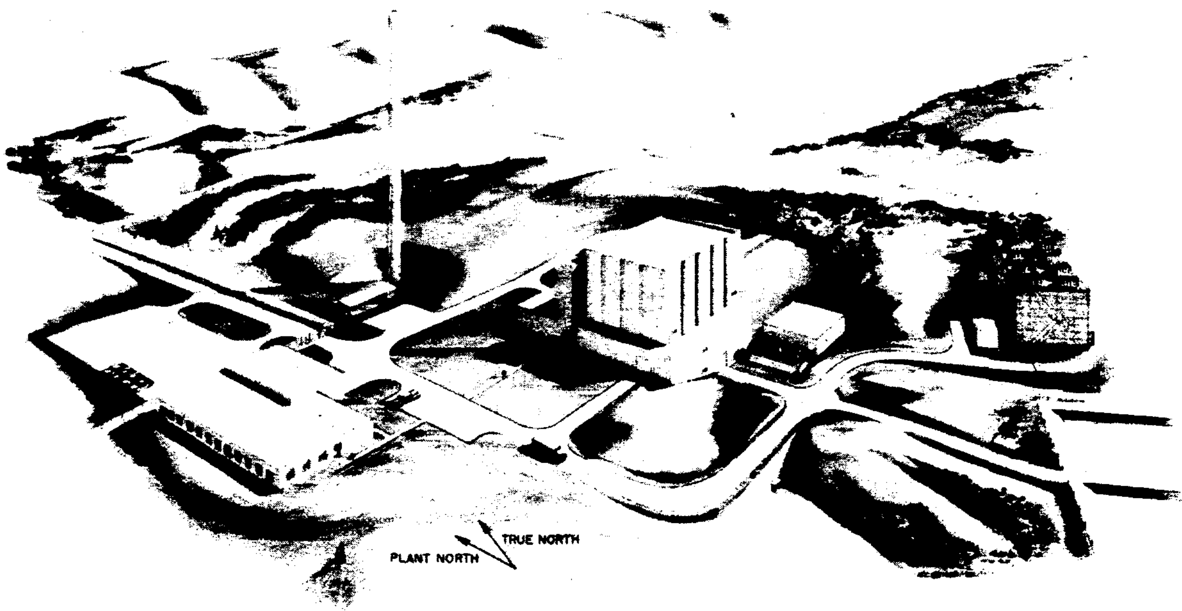
Volume 1A

Text

Edited by
F. T. Binford
T. E. Cole
E. N. Cramer

MAY 1968

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
U. S. ATOMIC ENERGY COMMISSION



View of the HFIR Site from Melton Hill

PREFACE

This description of the High-Flux Isotope Reactor (HFIR) 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 "Accident Analysis," which has been issued as ORNL-3573. The revised description presented here reflects the design as it was about January 1, 1968, at which time it does not appear probable that there will be any further significant modifications.

Volume 1A is basically descriptive in nature and is for the purpose of acquainting the reader with just what the HFIR is and how it operates. Except where necessary for clarity, little emphasis is placed on the design calculations. The interested reader will find in Appendix C a list of HFIR reports which constitute the detailed basis for the design. The various operating parameters cited are the actual values; however they may vary to some extent during a cycle or over a period of years.

Volume 2 contains a selected group of HFIR design drawings which, if used in conjunction with Volumes 1A (text) and 1B (figures), will materially aid in gaining an understanding of the system.

The information contained herein has been compiled from reports of studies performed by the various members of the HFIR project, which was directed by C. E. Winters until December 1961 and since then by A. L. Boch and T. E. Cole as Director and Technical Associate Director respectively. Among those who have contributed either by supplying descriptive material or by assisting in the review of the manuscript are A. M. Billings, F. T. Binford, T. G. Chapman, R. D. Cheverton, H. C. Claiborne, W. G. Cobb, T. E. Cole, B. L. Corbett, E. N. Cramer, J. W. Hill, N. Hilvety, J. E. Jones, R. V. McCord, H. A. McLain, J. R. McWherter, L. C. Oakes, T. H. Row, R. E. Schappel, J. H. Westsik, and their co-workers.

The Editors

CONTENTS

TABLES

1. INTRODUCTION

1.1 Historical Background and Motivation

On December 6, 1957, discussions were inaugurated at the Oak Ridge National Laboratory (ORNL) concerning the need for thermal-neutron fluxes an order of magnitude greater than were then available. As a result of the initial meeting, a series of informal seminars was conducted to explore further this need and to examine in some detail the technical problems associated with the design and construction of a reactor capable of producing such fluxes.[1] The primary conclusion reached was that the most pressing need for high thermal-neutron fluxes (i.e., 3 to 5×10^{15} neutrons cm^{-2} sec^{-1}) exists in connection with the production of transuranium elements and other isotopes.

The status of the transuranium production program was critically reviewed by the AEC Division of Research at a meeting on January 17, 1958. At that time it was decided to embark on a program designed to meet the anticipated needs for transuranium isotopes by undertaking certain irradiations in existing reactors. By late 1958 it became apparent that acceleration of this program was desirable. Accordingly, a meeting was held in Washington, D.C., on November 24, 1958; and, it was recommended[2] that a high-flux reactor be designed, built, and operated at ORNL with construction to start in FY 1961.

As a result of this decision ORNL, in March 1959, submitted a proposal to the U.S. Atomic Energy Commission (AEC). Authorization to proceed with the design of a high-flux reactor was received in July 1959, and a preliminary conceptual design of the reactor was published in February 1959.[3] This design was based upon the "flux-trap" principle, in which the reactor core consists of an annular region of fuel surrounding an unfueled moderating region or "island." Such a configuration permits fast neutrons leaking from the fuel to be moderated in the island and thus produces a region of very high thermal-neutron flux at the center of the island.*

Development of design criteria for the facility was begun by ORNL in 1959, and by March 1960 a general description of the proposed High-Flux Isotope Reactor (HFIR) was published.[4] The firm of Singmaster and Breyer was retained in March 1960 as architect-engineer for the purpose of handling the detailed design of the non-nuclear portions of the plant. Responsibility for design of the reactor core and the control and safety systems was retained by ORNL.

A review of the proposed reactor design[4] was made by the USAEC Hazards Evaluation Branch in April 1960 and by the USAEC Advisory Committee on Reactor Safeguards (ACRS) in May 1960. These reviews resulted in a recommendation that either conventional gastight containment be provided or that a location be sought which would provide some degree of isolation from the main body of ORNL employees. Site approval for construction of the reactor at a location in adjacent

*A bibliography containing references to the numerous studies of high-flux reactors is given in Appendix D.

[1] J. A. Lane *et al., Ultra High Flux Research Reactors*, ORNL-CF-58-7-117 (July 1958).

[2] J. H. Williams to A. M. Weinberg, letter dated Dec. 12, 1958.

[3] J. A. Lane, *High Flux Isotope Reactor Preliminary Design Study*, ORNL-CF-59-2-65 (February 1959).

[4] T. E. Cole, *High Flux Isotope Reactor – A General Description*, ORNL-CF-60-3-33 (March 1960).

Melton Valley was requested by ORNL in June 1960. After review at its 27th meeting (July 20–22, 1960) the ACRS concluded that the proposed reactor could be constructed at the Melton Valley location, with reasonable assurance that its operation would not entail undue risk to the health and safety of the public or of ORNL employees.

The architect-engineer completed the preliminary design and cost estimate (Title I) in February 1961 and the final building and non-nuclear equipment design (Title II) in January 1962. The H. K. Ferguson Company of Cleveland, Ohio, the general cost-plus-fixed-fee contractor for the Oak Ridge area, was awarded the construction contract; and in June 1961, preliminary construction activity was started at the site. In early 1965 construction was complete, and final hydraulic and mechanical testing was begun. Criticality was achieved on Aug. 25, 1965. The low-power testing program was completed in January 1966, and operational cycles at 20 and 50, 75, 90, and 100 Mw were begun.

1.2 Brief Description of the HFIR

The HFIR is a beryllium-reflected, light-water-cooled and -moderated, aluminum-clad fuel plate, flux-trap type reactor which utilizes highly enriched U^{235} fuel. The design power level is 100 Mw. Figure 1.2.1 is a cutaway view of the reactor, showing the major areas and important equipment items. A Summary Data Table is included in Appendix A.

The HFIR and its auxiliary facilities are located on the AEC reservation at Oak Ridge, Tennessee, at a point in Melton Valley approximately 1 mile south of the main site of ORNL (X-10).

The HFIR was designed primarily as a part of the overall program to produce transuranium isotopes for use in the heavy-element research program of the United States.

The reactor core, illustrated in Fig. 1.2.2, consists of a series of concentric annular regions, each approximately 2 ft high. A 5-in.-diam hole forms the center of the core. The target containing Pu^{242} and other transuranium isotopes is positioned on the reactor vertical axis within this hole. The fuel region is composed of two concentric fuel elements. The inner one, which contains 171 curved fuel plates, has an inside diameter of 5 in. and an outside diameter of $10\frac{1}{2}$ in. The outer fuel element contains 369 curved fuel plates and has inner and outer diameters of 11 and 17.134 in. respectively.

The fuel plates are 0.05 in. thick and are curved in the shape of an involute, thus providing a constant coolant channel width. The plates are of complex sandwich-type construction composed of U_3O_8-Al cermet held between covers of type 6061 aluminum. To minimize the radial peak-to-average power density ratio, the fuel is nonuniformly loaded along the arc of the involute. A burnable poison is included in the inner fuel element to further flatten the neutron flux and to reduce the negative reactivity requirements of the control plates. A typical core loading includes approximately 9.4 kg of U^{235} and 2.8 g of B^{10}. This provides a maximum of 11% available reactivity, and the average core lifetime is about 23 days.

The fuel region is surrounded by a concentric ring of beryllium reflector approximately 1 ft thick. This, in turn, is backed up by a water reflector of effectively infinite thickness. In the axial direction the reactor is reflected by water.

The control plates, in the form of two thin poison-bearing concentric cylinders, are located in an annular region between the outer fuel element and the beryllium reflector. These plates are driven in opposite directions by drive mechanisms located beneath the reactor. The inner control cylinder has its poison arranged so that reactivity is increased by downward motion. This cylinder is used for shimming and regulation; it has no fast safety function. The outer control cylinder consists of four separate quadrants, each having an independent drive and safety release mechanism. Reactivity is increased as the outer plates are raised. All control plates have three regions of different poison content designed to minimize the axial peak-to-average power density ratio throughout the core lifetime.

The reactor control and instrumentation system design reflects the very considerable emphasis placed on the importance of continuity of operation. Three independent safety channels are arranged in a coincidence system that requires agreement of two of the three for safety shutdowns. This feature is complemented by an extensive "on-line" testing system which permits the safety functions of any one channel to be tested at any time during operation. In addition to the independent safety systems, three independent automatic control channels are arranged so that failure of any one of the channels will not significantly disturb operation.

Three instrument thimbles near the core permit access for neutron- and gamma-sensitive instrumentation. Each thimble contains an ORNL multisection chamber. These chambers provide a gamma level, and an uncompensated neutron level for the safety system as well as a second uncompensated neutron level for the servo control system. Three servo-controlled fission chambers, located beneath the core, provide complete neutron flux information from shutdown to full power and thus eliminate the need for multiple range or overlapping range instruments. To aid in detecting any gross shifts in reactor power, a small stream of the primary water leaving each quadrant of the core is led past an N^{16} radiation detector. The temperature of the water leaving each core quadrant is also measured.

The reactor core assembly is contained in an 8-ft-diam pressure vessel, which is located in an 18-ft-diam cylindrical pool of water. The top of the pressure vessel is 17 ft below the pool water level, with the reactor center line $27\frac{1}{2}$ ft below the pool level. Adjacent to, and connected with, the reactor pool is a storage pool 20 ft deep, 18 ft wide, and $41\frac{1}{2}$ ft long. Underwater access is also provided to a small (8-ft-diam) pool intended for a future critical facility.

The primary coolant enters the pressure vessel through two 16-in. pipes above the core, passes through the core, and exits through an 18-in. pipe beneath the core. The flow rate is about 16,000 gpm, of which approximately 13,000 gpm flows through the fuel region and the remainder through the target, reflector, and control regions. The system is designed to operate at an inlet pressure up to 1000 psi; however, normal operation at 100 Mw requires an inlet pressure of only 600 psi. Under these conditions the inlet temperature is 120°F, the corresponding exit temperature is 164°F, and the pressure drop through the core is 110 psi.

From the reactor the coolant flow is distributed to three of four identical heat-exchanger circulation-pump combinations, each located in a separate cell adjacent to the reactor and storage pools. Each cell also contains a let-down valve which controls the primary coolant pressure. A secondary coolant system removes heat from the primary system and transfers it to the atmosphere by passing water over a four-cell induced-draft cooling tower.

Although the primary purpose of the HFIR is the production of transuranium isotopes, several other experimental facilities have been provided. These include (1) four horizontal beam holes which originate in the reflector; (2) four slant access facilities, called "engineering facilities," which are located adjacent to the outer reflector at an angle with the vertical; and (3) 38 vertical facilities of various sizes located in the reflector.

The reactor is housed in a poured concrete building approximately 128 × 160 × 86 ft high. The building is essentially airtight, with dynamic containment similar to that developed for the Oak Ridge Research Reactor being employed. Air is exhausted from potentially contaminated areas at a rate sufficient to assure that all leakage is inward. This air is passed through two absolute filters, a silver plated copper mesh filter and two charcoal beds before being released to the atmosphere through a 250-ft stack. Small volume releases of activity resulting from normal operation are handled through the hot off-gas systems, which are equipped with the same type of air cleaning devices as the main building exhaust system. Uncontaminated liquid effluent is discharged directly into a nearby stream. This flow is monitored, and automatic valves transfer it to retention ponds if a significant amount of radioactivity is detected. Contaminated liquid waste is pumped to the ORNL liquid-waste-disposal system from the retention ponds and from a hot-waste collection tank. Contaminated solid waste is collected and buried in accordance with the ORNL solid-waste-disposal procedures.

In addition to the reactor building, a 49- × 64-ft electrical building housing the switchgear and an office and maintenance building containing approximately 13,500 ft^2 of floor space are located at the HFIR site.

The site is supplied with 13.8 kv electrical power, 100 psi potable and fire-protection water, and 220 psi steam. All other materials and utility services are shipped in bulk or generated at the site. Two diesel-electric generators are available to furnish emergency power when required.

Oak Ridge National Laboratory, which is operated by the Nuclear Division of Union Carbide Corporation for the U. S. Atomic Energy Commission, is responsible for the operation of the HFIR.

2. REACTOR SITE

2.1 Location

The HFIR and its ancillary facilities are situated in the Roane County portion of the AEC reservation at Oak Ridge, Tennessee, and are shown in respectively increasing scale in Figs. 2.1.1 to 2.1.5. The HFIR is located in Melton Valley, south of the ORNL X-10 site. The facility is 1500 ft directly south of Building 7503 on the southerly slope of intervening high ground. Building 7503 now contains the Molten-Salt Reactor Experiment (MSRE) and was previously used for the Aircraft Reactor Experiment (ARE). Building 7500, the location of the original Homogeneous Reactor Experiment and Homogeneous Reactor Test, is 2000 ft northwest of HFIR. The Nuclear Safety Pilot Plant (NSPP) is currently located in this building. The Transuranium Processing Plant (TRU) is located adjacent to the HFIR. The main laboratory area of ORNL is 1 mile to the north, with Haw Ridge between it and the Melton Valley area. Melton Hill bounds the valley on southern side. The main access to the HFIR site is by a road leading from the east side of Building 7503.

The HFIR site is located within a well-established AEC-controlled area. The AEC Patrol covers the roads and adjacent area to restrict public access to certain designated routes through the controlled land. Suitable perimeter fencing is utilized to protect both the HFIR and the adjacent TRU facility. Adequate personnel and visitor control policies are followed so that only necessary operating personnel and persons having legitimate business are permitted within the exclusion area. Approximately 40 people are present in the reactor area 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 HFIR 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 HFIR site, as determined from the data in the 1960 census, is 147 persons per square mile. Table 2.2.1 lists the populations of the surrounding communities which have a population of over 2000 together with their distance and direction from the HFIR site. The rural population density in the four surrounding counties and in two other nearby counties is given in Table 2.2.2. A number of facilities are located within the AEC-controlled area; the approximate number of employees located in each plant is given in Table 2.2.3. These data indicate the total employment at each facility and do not attempt to show the breakdown according to shifts. However, it should be pointed out that practically all 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 $22\frac{1}{2}°$ sectors of concentric circles originating at the HFIR. Eight different distances were considered from the HFIR site: 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 2.2.4. 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.

Table 2.2.1. Population of the Surrounding Towns,[a] Based on 1960 Census

City or Town	Distance from HFIR (miles)	Direction	Population	Percent of Time Downwind	
				Night	Day
Oak Ridge	7	NNE	27,124	5.6	5.5
Lenoir City	9	SSE	4,979	4.3	6.0
Oliver Springs	9	N by W	1,163	2.3	2.7
Martel	10	SE	500[b]	1.4	2.8
Coalfield	10	NW	650[b]	0.5	1.1
Windrock	10	N by W	550[b]	2.3	2.7
Kingston	12	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	15	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	16	NE	4,943	11.6	9.0
South Clinton	16	NE	1,356	11.6	9.0
Powell	17	ENE	500[b]	8.3	6.8
Briceville	19	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	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	577[b]	9.5	6.1
Niota	30	SSW	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*, Addendum ORNL-CF-61-21-46, pp. 55, 56 (Aug. 14, 1961).

[b]1950 census.

Table 2.2.2. 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, J. H. Westsik, Addendum to ORNL-CF-61-2-46, *Molten-Salt Reactor Experiment Preliminary Hazards Report*, Addendum ORNL-CF-61-2-46, pp. 55,56 (Aug. 14, 1961).

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

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

Table 2.2.3. Number of Employees in Specific Oak Ridge Area (June 1963)

Area	Distance from HFIR (miles)	Direction		Total Number of Employees
HFIR				40
NSPP	0.4	WNW		6
MSRE	0.25	NNW		35
ORNL				3830
X-10 area personnel	0.75–1.25	NW	3327	
Construction personnel	0.75–1.25	NW	194	
7000 area personnel	1.0–1.4	NNE	309	
HPRR	1.1	ESE		12
Tower Shielding Facility	1.4	S		15
EGCR	2.0	NE		150
Melton Hill Dam[a]	2.25	S		
Construction personnel				
July 1963				350
December 1963				120
June 1964				25
Normal operation (remotely controlled)				2
K-25				2751
K-25 area personnel	5.0	WNW	2678	
Construction personnel			73	

Table 2.2.3 (continued)

Area	Distance from HFIR (miles)	Direction	Total Number of Employees	
Y-12	5.75	NNE		6866
Y-12 area personnel			5507	
ORNL personnel			909	
Construction personnel			450	
University of Tennessee Agricultural Research Laboratory	6.5	NE		160
Bull Run Steam Plant[a]	11.25	NE		
Construction personnel				
July 1963				1400
December 1963				1695
July 1964				1900
December 1964				1500
July 1965				1100
December 1965				700
Normal operation (one unit)				190

[a]Estimated from construction schedules, TVA, Knoxville, Tenn., June 5, 1963.

Table 2.2.4. Estimated Population Distribution

Radius (miles)	Sector															
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
0–0.5	0	0	0	0	0	0	0	0	0	0	0	0	0	6	0	35
0.5–1	29	0	0	0	0	0	0	0	0	0	0	0	0	0	789	1445
1–2	309	0	150*	0	0	12	0	0	15	0	0	0	0	221	738	18
2–3	0	0	0	24	0	41	20	0	90[a]	90	90	45	0	0	0	0
3–4	0	0	0	24	41	87	40	20	135	135	135	45	0	0	0	0
4–5	0	0	0	87	87	90	60	40	180	180	180	45	2,751	0	0	200
5–10	7944	20,428	460	7,706	7,706	7,706	783	6546	1567	1564	781	781	781	781	781	781
10–20	5320	13,318	6650[b]	56,414	55,914	23,131	6388	5660	4700	4700	1563	3573	16,741	2542	1500	4190

[a]Does not include Melton Hill Dam (see Table 2.2.3).
[b]Does not include Bull Run Steam Plant (see Table 2.2.3).
*Shutdown of the EGCR eliminates this group.

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-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.

(*a*) **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 of 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 –5°F and +103°F, recorded in December 1962 and September 1954 respectively. Table 2.3.1 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.

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

Table 2.3.1. ORNL Climatological Standard Normal Temperatures (1931–1960)

	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

(*b*) **Vertical Temperature Gradient.** – Information on the temperature gradient, frequency, and mean wind speed for each month are 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. 2.3.1.

(c) **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 maximum monthly precipitation occurs normally in March and has a value of 5.44 in.

The average monthly precipitation is given in Table 2.3.2.

Table 2.3.2. 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

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 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 maximum 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 short duration.

(*d*) **Wind.**[4] – The valleys in the vicinity of the HFIR site are oriented northeast-southwest, and considerable channeling of the winds in the valley occurs. This is evident in Fig. 2.3.2, which

[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, Tenn., Area Station (X-10).

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

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 upvalley from southwest and west-southwest approximately 40% of the time, with a secondary maximum of downvalley 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 represent 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 marked 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 (1956–1960) of data are shown in Figs. 2.3.3 and 2.3.4 (Ref. 2). These represent the wind direction and frequency and percent calm under inversion and lapse conditions for the ORNL area and are applicable to the HFIR location.

Considerable variation is observed both in wind speed and direction within small distances in Bethel Valley and Melton Valley. In general, at night or under stable conditions the winds tend to be northeast or east-northeast and rather light in the valley, regardless of the gradient wind. However, strong winds aloft will control the velocity and direction of the valley winds, reversing them or producing calms when opposing the local drainage. During the day, the surface winds tend to follow the winds aloft, with increasing reliability 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 the well-developed stable situation, however, a very light air movement will follow the valley as far downstream 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 a valley location, wind direction will be governed by the local valley wind regime and the degree of coupling with the upper winds.[4]

The wind flow between Melton Valley and the X-10 area was investigated in conjunction with the ARE hazards analysis.[4] The information is included here because of the close proximity of the HFIR to the ARE site, the present site of the MSRE.

Two patterns of wind flow appear to be of particular significance: from the 7500 area northwest over Haw Ridge to the X-10 area and from the 7500 area west to White Oak Creek, then northwest through Haw Gap, and finally north to the X-10 area.

The frequencies with which these patterns occurred from September to December 1950 were normalized to the 1944 to 1951 wind record at the X-10 area by a ratio method. The normalized frequencies are shown in Table 2.3.3.

Table 2.3.3. Frequency of Wind Patterns Between Bldg. 7500 and X-10 Areas

	Frequency of Wind Pattern (%)	
	Over Ridge	Through Gap
All observations	2.5	0.4
Day (9 AM to 5 PM)	4.3	0.6
Night (9 PM to 5 AM)	0.0	0.4
Light wind (1 to 4 mph)	2.6	0.4
Stronger wind (5 mph and over)	2.7	0.3

A comparison of pibal (pilot balloon) observations made throughout 1949 to 1950 at Knoxville and Oak Ridge shows that above about 2000 ft the wind roses at these two stations are almost identical. This similarity of data makes possible the use of the longer period of record (1927 to 1950) for Knoxville and tends to eliminate the abnormalities introduced by the use of the short record at Oak Ridge.

Annual wind roses are shown for Knoxville (1927 to 1950) and Nashville (1937 to 1950) in Fig. 2.3.5. Since pibal observations are made only when no low clouds, dense fog, or precipitation are present, they are not truly representative of the upper wind at all times. Three years of rawin (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. A comparison of these wind roses for Knoxville and Nashville shows that the mode for winds above 3000 m MSL should be shifted to the west instead of west-northwest when observations with rain are included in the set.

The northeast-southwest axis of the valley between the Cumberland Plateau and the Smoky Mountains continues to influence the wind distribution over the Tennessee Valley up to about 5000 ft, although the variations within the Valley do not extend above about 2000 ft. Above 5000 ft, the southwesterly mode gives way to the prevailing westerlies usually observed at these latitudes.

Previous investigation of the relation of wind direction to precipitation indicates that the distribution of direction is very little different from that of normal observations.[4] This is consistent with the experience of precipitation forecasters that there is little correlation between surface wind direction and rain, particularly in rugged terrain. Figure 2.3.5 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 any given level is shifted to the southwest or south-southwest from from west or southwest, and the velocity is somewhat higher during the occurrence of precipitation, with the shift being most marked in the winter.

Tornadoes rarely occur in the valley between the Cumberlands and the Great Smokies. It is highly improbable that winds greater than 100 mph would ever occur at the HFIR site.

(e) **Atmospheric Diffusion Characteristics.** – The method proposed by Pasquill[5] is widely used to calculate the dispersion of airborne wastes 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 2.3.4 gives the frequency of these various conditions at the HFIR site.

A study[6] has been made of the average duration of inversions in the Oak Ridge area, which indicates 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 2.3.5.

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. 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.

(f) **Environmental Radioactivity.** – Data on the environmental levels of radioactivity in the Oak Ridge area[7] are given in Tables 2.3.6 to 2.3.9.

2.3.2 Geology and Hydrology

The bedrock beneath the HFIR site is a dark-gray calcareous clay shale with a bearing value of 6 tons/ft^2, when unweathered. Overburden on top of this fresh unweathered shale averages 20 ft in

---

[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 2.3.4. Atmospheric Stability Constants at Oak Ridge[a]

Condition	Occurence (%)
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 2.3.5. 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

thickness and consists of a thin blanket of organic topsoil generally less than 1 ft deep on top of weathered shale. The latter, if confined, has a bearing value of 3 tons/ft^2. The strata of the shale are highly folded and faulted, and the dip, although it averages about 35° toward the south, is irregular and may vary from horizontal to vertical. Melton Valley, in which the HFIR is located, is underlain by the Conasauga shale of the Middle and Upper Cambrian Age. The more-resistant rock layers of the Rome formation, steeply inclined toward the southwest, are responsible for Haw Ridge, which parallels the valley immediately to the northwest. These layers dip beneath the shales of the Conasauga group in Melton Valley. The shale layers in the area are in keeping with the general structure of the surrounding region, as reported in a geological survey of the area.[8] Conasauga shale is a dark-red shale containing thin layers and lenses of limestone that are generally irregular in distribution. However, there are no persistent limestone beds in the upper strata of the shale layers and, consequently, no underground solution channels or caverns to permit rapid and free discharge of water underground.

The dominantly clay soils of the Oak Ridge area are generally of low permeability, so that the surface runoff of water is rapid. Observations in test wells show that the Conasauga shale, although relatively impermeable, is capable of transmitting small amounts of water through the soil a distance of a few feet per week. However, all the radioactive isotopes, except ruthenium, apparently become fixed near the point of entry into the soil. It is concluded that groundwater flow in the soil

[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).

Table 2.3.6. 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 2.3.7. Radioparticulate Fallout – 1962

Averaged weekly from gummed paper data

Station No.	Location	Long-Lived Activity ($\mu c/cc$)	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

surrounding the site will be small and slow (a few feet per week) and that natural chemical fixation will reduce the level of the activity of mixed, nonvolatile fission products by more than 90%.

The Conasauga shale formation is an extensive formation which is quite heterogeneous in structure and relatively low in permeability.[8] The depth and extensiveness of the formation provide a large capacity for decontamination of any radioactive liquid reaching it, and the slow rate of percolation improves the efficiency of the decontamination by ion exchange and radioactive decay. However, the heterogeneity of the formation makes it difficult to predict the pattern or rate of water and radioactivity movement. Most of the seepage is along bedding planes parallel to the strike. The drainage pattern of Melton Valley is shown in Fig. 2.3.6. The smaller streams draining Haw Ridge and Copper Ridge all terminate in Melton Branch, which enters White Oak Lake after merging with White Oak Creek. Several other smaller streams enter White Oak Creek at different points along the shoreline. All drainage from the HFIR will ultimately reach White Oak Lake. Release from this lake is continuously monitored, so that adequate control of any released material is possible. Drainage from the HFIR site is in three directions. The two small streams on the east and west of the site, as shown in Fig. 2.3.6, receive water from surface runoff and roof drainage. The stream to the east of the site enters Melton Branch 300 ft south of the HFIR waste-retention ponds. The other stream enters Melton Branch 1100 ft west of the ponds. Ground contours and plant drainage eliminate any possible flooding of these ponds by storm water drainage, as shown on Singmaster and Breyer drawing No. 1546-05-U-7101, Area Utility Plan, South Part, in Vol. II.

Table 2.3.8. 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 2.3.9. 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).

Storm drainage will also be discharged directly south into Melton Branch. This water originates from surface runoff and roof drainage of the south side of the reactor and electrical building. It combines with the cooling tower blowdown and the flow from the process waste pond before discharging into Melton Branch through a connecting ditch. The storm drainage system is based on a runoff coefficient of 30% for unpaved or seeded areas. The high runoff coefficient applied to the unpaved areas is justified by the highly impervious characteristics of the soils.

In addition to expected rainout of airborne waste, there are other sources of contaminated groundwater flow from the HFIR. The HFIR process waste system is designed to receive liquid wastes, other than sanitary and storm system drainage, having small amounts of radioactivity. Flow from the process waste system will normally be to a holdup pond which provides an opportunity for sampling as well as settling and decay time for short half-life fission products, thus permitting the effluent to be discharged into Melton Branch. Process waste which has a significant level of contamination is automatically diverted to a second retention pond, from which it is pumped to the ORNL liquid waste system for treatment and disposal. The retention ponds are excavated in a shale area where seepage is not of importance because of the ion exchange properties of the shale. The ponds are not lined; however, the embankment slopes are stabilized against erosion with a layer of crushed stone.

Cooling tower blowdown, approximately 240 gpm, will normally bypass the process waste system through the storm and sanitary drain system to Melton Branch. The activity level of the blowdown is monitored as it leaves the cooling tower area. Should this activity exceed the low-level set point, the flow will be diverted to the process waste system.

The intermediate level waste (ILW) is stored in a 13,000-gal tank, from which it is pumped to the ORNL liquid waste system for treatment and disposal.

Downstream from ORNL numerous uses are made of the Clinch River water by both municipalities and industries, as shown in Table 2.3.10. The first downriver water consumer is the K-25 plant, whose water intake is located at river mile 14.4. Representative flows of the Clinch, Emory, and Tennessee Rivers are shown in Table 2.3.11. The local flow pattern in the Watts Bar reservoir during the period May to September is profoundly modified by the differences in water temperature between the Clinch River and Watts Bar reservoir. When the Clinch River is considerably cooler, stratified flow conditions due to density differences may exist (the cooler water flowing on the bottom beneath the warmer). This phenomenon markedly affects the travel time of water through the reservoir and complicates the analysis of flow. In addition, during the period of stratified flow, some Clinch River water may flow up the Emory River as far as the Harriman water plant intake.

There is very little probability of groundwater contamination in areas removed from the ORNL site. There are no developed groundwater resources in the area, due to the fact that formations present are for the most part too impermeable to hold or transmit any significant quantity of water.

2.3.3 Seismology

Information on the frequency and severity of earthquakes in the East Tennessee area is reported in the ART Hazards Summary Report.[9] Earthquake forces generally have not been 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.

---

[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 2.3.10. 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.

Table 2.3.11. 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.

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-Neuman 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."

3. BUILDINGS

3.1 Introduction

Three principal buildings are located at the reactor site. These are the reactor building, the office and maintenance (O and M) building, and the electrical building. Another large building, the Transuranium Processing Plant, is approximately 250 ft north of the reactor building. In addition, several small buildings, including the cooling tower equipment building and a shed to house the stack fans, are located nearby. The area is suitably fenced and is a controlled-access area. The general layout is illustrated in Fig. 2.1.5 and the frontispiece. Figures 3.1.1 and 3.1.2 illustrate the equipment arrangement within the reactor building.

3.2 Reactor Building

The reactor building contains approximately 57,200 ft^2 of floor area and has a total volume of approximately 1,329,000 ft^3, as determined by AEC criteria.[1] It measures approximately 128 × 160 ft at the ground floor and extends from an elevation of 793 ft at the subpile room floor to an elevation of 903 ft 9 in. at the reactor bay roof. Construction conforms to the provisions of the Southern Building Code[2] and the applicable requirements of the National Fire Protection Association.[3]

The main portion of the reactor building is enclosed by a poured concrete envelope within which lie the reactor bay, experiment room, beam room, and the primary coolant loop. The remainder of the building, which is located outside the envelope and on the north side of it, is constructed of concrete block. Since this wing contains the water treatment equipment and control room, it is referred to as the "water and control wing" or simply the "water wing." The general arrangement is shown in Figs. 3.2.1 to 3.2.6.

The main building has floors at elevations of 817 ft (ground floor), 833 ft (first floor), and 851 ft 2 in. (second floor). The second floor has a high bay structure to accommodate the 50-ton crane which serves the reactor. The reactor building is located on a slope between original ground elevations 850 and 820 ft, thus permitting truck access to each of the three major levels.

The first three floors in the water wing correspond to those in the main building; however, a third floor at elevation 865 ft 2 in. contains the reactor control room.

3.3 Envelope Construction and Containment

Those portions of the reactor system which contain significant quantities of radioactive material are enclosed within a reinforced concrete envelope capable of containing fission product releases. Specifically, the envelope contains the reactor bay, experiment areas, heat exchanger cells, and pipe tunnels. The major areas are isolated from each other and are ventilation-controlled

[1] "AEC Manual of Instructions," Chap. 6302, Subsection 17.

[2] "Southern Standard Building Code," Southern Standard Building Code Congress, 1960–61.

[3] "Standards of the National Board of Fire Underwriters," National Fire Protection Association, pamphlets such as: NBFU 72, June 60, *Protective Signaling System;* NBFU 14, Oct. 52, *Standpipe and Hose Systems;* and NBFU 13, June 61, *Sprinkler Systems.*

to reduce the possible spread of contamination. Concrete surfaces have smooth finishes and are coated with acid-resistant paint to facilitate decontamination.

The walls of the main building are of 8-in.-thick reinforced concrete, and the roof is of $5\frac{1}{2}$-in.-thick reinforced concrete. The walls and roof are designed to withstand an internal pressure of 0.5 psi above the outside pressure. Joints are protected by stops. Utility penetrations, such as pipes, conduits, and louvers, are sealed to the envelope in such a way as to also withstand a differential pressure of 0.5 psi around the openings. Doors and dampers which will maintain an internal pressure of 0.5 psi can be installed but are not included in the present design.

Separation between different floors is provided by concrete floor slabs and by suitable doors at the stairways. Doors leading to the outside or to areas external to the containment area are fitted with door closers and are weather-stripped to control leakage of air. These doors will be normally closed during reactor operation.

An observation gallery, which overlooks the reactor bay, is located just south of the control room (see Fig. 3.2.6). This gallery is provided with fixed windows in the wall of the containment envelope which permit a view of the activities in the reactor bay. These windows are also designed to withstand a differential pressure of 0.5 psi.

An air lock type of personnel entry is provided at the north wall of the reactor bay. An emergency air lock exits to the roof above the first-floor extension at the southwest corner of the reactor bay. The truck entrance on the northeast corner of the reactor bay is an air lock designed to accommodate a 65-ft-long low-boy tractor-trailer. These air locks do not provide for absolute sealing but are designed to ensure in-leakage of air consistent with the containment requirements (see Sec. 4).

3.3.1 Reactor Bay (Second Floor, Main Building)

The main operating floor at elevation 851 ft 2 in. surrounds the reactor pool (see Figs. 3.3.1, 3.2.5, and 3.2.6). Immediately above is the high bay area in which is located a 50-ton traveling bridge crane. The top of the crane track is at elevation 889 ft 6 in., giving an 8-ft 9-in. clearance below the roof. Hook travel extends to 5 ft 9 in. from the north wall columns, to 8 ft 6 in. from the south wall columns, and to 9 ft 1 in. and 10 ft 6 in. from the east and west walls respectively. The vertical travel of the crane hook extends from 37 ft above to 19 ft below the 851 ft 2 in. floor level.

The $5\frac{1}{2}$-in.-thick roof slab is covered by insulated, built-up roofing. This roof and the others on the building are designed to withstand a live load of 20 psf. The reactor bay roof is 79 ft wide by 110 ft long.

Located in the reactor bay are the personnel bridge spanning the reactor and storage pools, equipment access hatches for the primary pumps and heat exchangers, six chilled-water air-conditioning units, and a stainless steel decontamination pad. The area in the northeast corner between column lines 5 and 6 is reinforced to serve as a loading area for trucks entering at the air lock in the east wall. A 50-ton truck load can be supported in this area.

The decontamination station is located at the northwest corner of the reactor bay. This station, which is provided primarily for decontamination of carriers, is 22 ft long by 19 ft 6 in. wide. It consists of a curbed stainless steel drain pad with provisions for temporary enclosure if required during decontamination operations. The pad is fabricated of 304 stainless steel. The floor supporting the decontamination pad is designed to support a load of 50 tons on 9 $ft.^2$

Elsewhere in the reactor bay the reinforced concrete slab floor is designed for a live load of 500 psf plus an additional 10-ton movable load on 6 ft^2. Floors are level and provided with floor drains, both near the columns and at the pool perimeter.

The area immediately above the pools is open except for the three movable 18-ft-wide by 10-ft-long transparent plastic pool covers and the personnel bridge which traverses the length of the pool. The bridge moves on rails supported in the pool structure. The floor of the bridge is at elevation 851 ft 2 in., flush with the second floor. The floor west of the reactor pool has a recessed area to accommodate one half of the pool cover. This recess is covered with steel plate overlaid with vinyl-asbestos tile. Removable sections of the steel plate provide access to an experiment facility hatch. The water level of the pool is normally at elevation 848 ft, approximately 3 ft below the floor of the personnel bridge and the reactor bay floor.

Entrance into the area is provided through a personnel decontamination room which also serves as an air lock. It is located in the north wall between column lines E and F. Two emergency personnel exits, which are normally closed during reactor operation, open into the corridor behind the north wall. An emergency-exit personnel air lock leading to the roof of the first floor is located in the southwest corner. A door is provided from this roof into the water wing.

3.3.2 Experiment Room (First Floor, Main Building)

Space is provided for experimental work in the experiment room located at elevation 833 ft to the west and south of the reactor pool structure directly below the main operating area (see Figs. 3.3.2, 3.3.3, 3.2.2). The ground and first floors are extended 17 ft to the west and 15 ft to the south to provide this additional working space.

The oblique experiment facilities which penetrate the biological shield around the reactor pool and the vertical access holes to the shutter region of the beam holes terminate in the experiment room. This shown in Fig. 3.3.3. It was not planned to install the slant experiment tubes initially, so the access penetrations were blanked off and shielded. Vertical access holes to the beam tube shutter region are shielded, when not in use, by concrete plugs encased in carbon steel and by stacked concrete block. Blanked-off 2-in. pool water inlet and return lines and an intermediate-level-waste tap are provided at service stations adjacent to the slant tube locations. Process waste floor drains are also provided at these locations. The primary coolant flow, temperature, and pressure monitoring devices are located in this room and extend along the west wall adjacent to the primary pump and heat exchanger cells. Also located in the experiment room is the hot-water-injection system for testing the heat power temperature sensors.

The experiment room may be entered through a personnel door located in the north wall between column lines A and B which permits access to the first floor of the water wing. The stairway located in the south wall between column lines E and F allows access to the ground floor. In addition, there is a 10-ft truck door opening to the outside located in the west wall between column lines 8 and 9. A personnel door is located directly south of this truck door. Although these doors into the experiment room are equipped with closers and are normally closed when the reactor is operating, they may be used during operation.

Exposed structural steel is used for framing the reactor bay floor (the experiment room ceiling) and the experiment room floor. This will facilitate future programs by making it possible to readily attach experimental equipment to the structural members in the experiment room.

Floors in the experiment room are designed to withstand live loads of 750 psf. Static loads from the removable shielding blocks located around the pool wall are borne by the concrete beam hole shield below.

3.3.3 Beam Room (Ground Floor, Reactor Building)

The beam room is located on the ground floor of the reactor building at elevation 817 ft and immediately beneath the experiment room (see Figs. 3.3.4 and 3.2.5). This area contains the ports for the horizontal beam tubes and thus is located approximately on the same level as the reactor core. The general arrangement of these experiment facilities is shown in Fig. 3.3.5.

The radial beam tube, oriented in the east-west direction, provides for time-of-flight experiments. It is arranged to permit the future construction of a 400-m flight tube beyond the face of the building wall. A 48-in.-diam concrete pipe, which will enclose the flight tube, is installed on the west side of the building. It extends from the building wall to the HFIR perimeter fence.

Each beam hole is designed to be equipped with a barrel shutter and a high-density-concrete beam hole plug. Rails covered with removable steel plate are provided at each port to facilitate movement of future beam hole shields. During initial operation without experiments, concrete blocks will be placed in the beam hole cavity. A connection to the hot off-gas system is provided at each beam port. Appropriate services and utilities are led directly to the cavity adjacent to the beam hole.

The beam room is normally entered from personnel doors leading from the water wing. The stairwell located on the south wall between column lines E and F allows access to the first floor. Direct access from outside is provided by a 10-ft truck door and a personnel door located in the south wall adjacent to column line E. The doors into the beam room, except for the truck door, are equipped with closers and are normally closed when the reactor is operating. They may; however, be used during operation.

Two equipment rooms located in the southeast corner of the ground floor house the pony motor batteries (see Sec. 10) and a portion of the electrical equipment.

Floors in the beam room are designed for a live load of 2500 psf plus rolling load capacity adequate to handle the heavier equipment such as the beam hole shielding. The allowable floor load in the equipment rooms is 200 psf. Floor slabs at ground level are underlain by a polyethylene sheet vapor barrier.

As in the case of the experiment room, exposed structural steel framing facilitates the installation of experimental equipment.

3.3.4 Subpile Room

The subpile room, located directly beneath the reactor, houses the reactor control plate drive mechanisms and the N^{16} monitoring system. The arrangement and location can be seen in Figs. 3.2.1 and 3.2.5. A work platform is provided to facilitate inspection and maintenance of the equipment installed in this location. The floor and walls of the subpile room are formed of 4-ft-thick reinforced concrete. A stairway and equipment hatch from the beam room provide access to the subpile room.

Although the subpile room may be entered during operation for inspection of equipment, the radiation background ($\sim$10 mr/hr) is higher than that in the rest of the building. It will not be continuously occupied while the reactor is operating.

3.3.5 Primary Pump and Heat Exchanger Cells

The pumps and heat exchangers for the reactor primary cooling system are located in four rectangular cells (see Fig. 6.2.4) adjacent to the north wall of the reactor pool. The cells extend vertically from elevation 793 ft to the reactor bay floor at elevation 851 ft 2 in. The pump bases are located at elevation 826 ft. Three of the cells have a square cross section 11 ft 6 in. on a side and the fourth has a cross section of 11 ft 6 in. by 16 ft × 9 in. In addition to the heat exchanger and pump, the larger cell contains the flow measuring element for the secondary coolant. The cells are provided with floor drains at elevations 793 and 826 ft.

Personnel access to each of these cells is provided from the north side by means of shielded labyrinths and rubber gasketed doors located at elevation 838 ft 4 in. (see Figs. 3.2.2, 3.2.6, and 4.2.2). These doors have adjustable, gravity-operated backdraft dampers to prevent reverse airflow and to assist in pressure control of the cells. Platforms in the cells at elevations 826 and 809 ft are reached by ladders located in the cells.

A separate hatch in the reactor bay floor is provided for each heat exchanger and each pump. The hatches are of reinforced concrete 5 ft 6 in. thick. The heat exchanger hatches measure 7 ft

2 in. by 6 ft 4 in. at the top and step down to 6 ft 2 in. by 5 ft 4 in. at the bottom. The pump hatches are 5 ft 11 in. square at the top and 4 ft 11 in. square at the bottom.

The pipe tunnel containing the primary coolant lines is located just north of the heat exchanger cells at elevation 818 ft (see Fig. 3.2.1).

3.4 Control and Water Wing

The control and water wing is located just north of the main portion of the building outside of the poured concrete envelope. This section of the building is of concrete frame construction with concrete floors and roof. The exterior walls are concrete block with wire reinforcement in the horizontal joints. The floors in this section of the building correspond in elevation to those within the main building. In addition, there is a third floor at elevation 865 ft 2 in. for the reactor control room.

An elevator and stairwell in the southwest corner provide access to all four floors. The stairwell emerges onto the roof above the third floor. A second stairwell is located in the southeast corner. The elevator which serves the four floors travels at 75 fpm under full load capacity of 10,000 lb. Overall dimensions of the elevator platform are 6 ft 9 in. by 7 ft. It is capable of carrying a concentrated rolling load of 10,000 lb on a dolly with wheel spacing 4 ft wide by 6 ft wheelbase. The elevator is controlled by push buttons in the car.

3.4.1 Reactor Control Room and Amplifier and Relay Room

The control room is located on the third floor at elevation 865 ft 2 in. Also located at this level are the observation gallery, two offices, toilet facilities, and rooms for parts and instrument storage. The general arrangement is shown in Fig. 3.4.1. A stairway between the amplifier and relay room and control room is located at the center of the north wall.

A 66-ft-long observation gallery extends along the south wall and overlooks the reactor bay. The fixed glass panels which penetrate the main building envelope are designed to withstand a differential pressure of 0.5 psi. The gallery is separated from the control room by a glass wall panel, and control room operations can be viewed from the observatory gallery. Visitor traffic is routed into the building through the northwest door, located on the first floor of the water wing, and up the west stairs or the elevator to the observation gallery.

The amplifier and relay room, which contains most of the electronic equipment, relays, and recorders collecting information not requiring display in the control room, is located on the second floor at elevation 851 ft 2 in. directly beneath the control room. Also located at this elevation are the instrument-battery room, a health physics office, toilet facilities, and locker room. A personnel decontamination room extends into this area on the south wall between column lines E and F. This room, however, is within the poured concrete envelope and serves as the main personnel entrance to the reactor bay.

3.4.2 First-Floor Control and Water Wing

At elevation 833 ft the water wing widens to an area measuring 127 ft by 82 ft. Housed on this level (the first floor) are the heating and ventilating equipment, the pool equipment, the fire alarm equipment, and a number of plant utilities as shown in Fig. 3.4.2. Shielded cells for the pool deaerator, the pool demineralizer, the pool coolant filter, and the primary coolant deaerator are located on this floor. Access to the primary pump and heat exchanger cells is also provided in this area via stairs to a platform located along the south wall.

3.4.3 Ground-Floor Control and Water Wing

The primary and pool cleanup equipment cells continue down to the ground floor at elevation 817 ft. This floor also contains a large equipment area along the north end and a staff shop and toilet facilities at the southwest corner, as shown in Fig. 3.4.3. A 14 ft by 8 ft 6 in. equipment-removal hatch extends out from the north wall of the building. A 7-ft 6-in.-wide equipment-removal corridor extends along the south wall of this wing. Accesses to the pipe tunnel and to the electrical manhole are provided at the east end of this corridor.

3.5 Auxiliary Buildings

3.5.1 Electrical Building

The electrical building is of masonry construction with 3136 ft^2 of gross floor space which contains a switchgear battery room, two diesel generator rooms, and a high-voltage switchgear room (see Fig. 3.5.1).

The floor is made of 6-in. slab concrete poured upon a 6-in. layer of crushed stone. The battery room floor has a vinyl coating on top of the concrete. The roof is made of 3-in. light concrete covering Corruform roof decking. Walls are 8-in.-thick concrete block.

A roof fan ventilates the battery room and switchgear area. Two louvered penthouses are provided (one in each diesel generator room) to exhaust cooling air pulled in through combination fixed and automatic louvers by the diesel engines.

An acid-proof sink and safety shower are provided in the battery room.

3.5.2 Cooling Tower Equipment Building and Fan Shed

The cooling tower equipment building is of masonry construction with a total of 640 ft^2 of floor space. The floor plan is shown in Fig. 3.5.1. Of this, approximately 310 ft^2 is occupied by electrical equipment, including the cooling tower fan and pump controls. The remaining area is utilized by the cooling tower sprinkler system dry-pipe valves, and by the chemical treatment equipment and chemical storage. This includes dual acid metering pumps, an acid mixer, a chromate-phosphate mixing tank, and dual chromate-phosphate metering pumps. The roof of the cooling tower equipment building is constructed of lightweight concrete-filled corrugated steel decking.

The fan shed is an open, roofed structure except for a 13- by 8-ft area which is enclosed to house the fan control instrumentation. It provides 1420 ft^2 of floor space under a transite roof. Provision is made for shielding the hot off-gas fans and ducts to a height of 6 ft 8 in. by stacked solid concrete block 2 ft thick. The building is located just southwest of, and adjacent to, the stack.

3.5.3 Office and Maintenance Building

The office and maintenance building, shown in Fig. 3.5.2, is a one-story masonry structure. It is approximately 13,500 ft^2 in gross area, including 4200 ft^2 of shop area. The office block contains 21 offices, a conference room, a lunchroom with space for vending machines, a locker room, toilet facilities, and a building facilities room.

Cooling and heating systems for all air-conditioned rooms except the lunchroom consist of packaged fan coil units with fresh-air intakes. Approximately 75% of the air is recirculated. The lunchroom is equipped with a self-contained package air-conditioning unit having a capacity of $7\frac{1}{2}$ tons.

4. CONTAINMENT, VENTILATION, AND AIR CONDITIONING

4.1 Introduction

The air and gas handling facilities at HFIR have been specifically designed to minimize the spread of contamination inside and outside the building in the event of an accidental release of activity and to provide maximum safety during normal operation. Two separate systems (Fig. 4.1.1) are provided for the disposal of gaseous waste: the special building hot exhaust system (SBHE), which provides dynamic containment in the event of an abnormal release of activity, and the hot off-gas system (HOG), which handles the routine disposal of gaseous activity from the various system components. Certain areas, such as the control room and offices which are normally clean areas, are isolated from the other portions of the building and are served by a separate air-conditioning system.

The SBHE system is designed to provide a constant flow of air through those portions of the building which contain components capable of releasing significant quantities of activity. These areas include the primary heat exchanger cells and pipe tunnel, the reactor bay, the primary coolant deaerator and demineralizer cells, and the primary coolant demineralizer pump cell. Certain other areas, including the beam room, experiment room, and the cells containing the pool cleanup equipment, are also served by the SBHE system because of the configuration of the building and convenience in locating the equipment. By maintaining appropriate pressure gradients, clean air flows into the areas served by the SBHE system, then through ducts to appropriate filtering equipment, and finally into the atmosphere from the top of a 250-ft stack.

Although the HOG system is actually a gaseous-waste-disposal system, its description is included in this section for convenience, because it is very similar to the SBHE in arrangement, components, and operation. An understanding of this system will provide background information for the material presented in succeeding sections.

The HOG system is designed to handle low-volume high-concentration releases which normally accompany operations with radioactive material. It is connected directly to the components (deaerators, demineralizers, etc) which release the gaseous activity. There are actually two such systems: the closed hot off-gas system (CHOG), which collects gas from components which may be pressurized, and the open hot off-gas system (OHOG), which collects gas only from unpressurized components. Both the CHOG and OHOG systems discharge through filters to the stack.

Both charcoal and absolute filters, designed to reduce the potential activity in the effluent streams to tolerable discharge levels, are included in the filter assemblies of the SBHE and HOG systems. Silver-plated copper-mesh filters are located just upstream from the SBHE and HOG charcoal filters. The primary purpose of these silver filters is to retain the bulk of the iodine, thus preventing an excessive heat load in the charcoal.

4.2 General Containment Philosophy

The general concept of dynamic containment has been adequately described elsewhere.[1] Briefly it consists in maintaining an inward leakage of air into the contained region by creating

---

[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).

a partial vacuum in that region. The air exhausted from the contained region is decontaminated by suitable filtering equipment and is discharged to the atmosphere in a manner designed to eliminate excessive environmental pollution.

In the HFIR this concept has been applied and modified slightly to obtain better control of internal air movement. The building as a whole is under dynamic containment. In addition, the flow of air within the building is always from an area of low contamination potential into an area of greater contamination potential. For convenience, the movement of air will be described in terms of pressure differentials; however, it should be clearly understood that the important parameters are the direction and flow of the air, not the measured pressure differentials which may or may not be a true indication of flow. Moreover, it must be realized that leakage – controlled leakage – of air from outside the building and from certain designated regions in the building is necessary if the system is to function properly.

For containment purposes the building is divided into four major ventilation-control areas or zones as follows (Figs. 4.2.1 to 4.2.4):

Zone 1 operates under slight positive pressure with respect to the ambient atmospheric pressure; it includes the control rooms, offices, and other facilities which are normally occupied and which contain no significant potential sources of activity. Zone 1 includes the second and third floors of the control and water wing.

Zone 2 is a generally clean area but operates at a slightly negative pressure (approximately −0.1 in. WG). It includes the process equipment rooms on the ground and first floors of the water wing.

Zone 3 includes all the area inside the main reactor building with the exception of the primary heat exchanger cells and the pipe tunnel. It also includes those portions of the water wing that contain the primary and pool cleanup cells. It is directly connected to the SBHE system and is designed to operate at negative pressures up to 0.3 in. WG.

Zone 4 includes the primary heat exchanger cells and the main pipe tunnel. It too is connected directly to the SBHE system and designed to operate at pressures slightly lower than those for zone 3.

Two rooms in the southeast corner of the ground floor (G-12 and G-13, which contain electrical equipment and batteries) require ventilation only by supply and exhaust fans. They are not considered to be within the contained area and are maintained at approximately atmospheric pressure.

Zone 1 is self-contained from a ventilation and exhaust standpoint. It is equipped with a large chilled-water air-conditioning unit which maintains a slight positive pressure with respect to the outside atmosphere.

Zone 2 has its own supply and exhaust fans. The desired pressure in zone 2 is maintained by regulation of the amount of air supplied and control of leakage into adjacent zones 3 and 4.

The negative pressure in zone 3 is maintained by the operation of various individual supply dampers. A manually operated volume control damper located in the main SBHE exhaust duct is periodically adjusted to maintain a fixed volume of exhaust air. The volume of intake air (controlled inward leakage), which enters through the fresh-air intakes on the various air-conditioning units, is controlled by pressure-regulated dampers to maintain the required negative pressure in the area. The fresh-air intake regulators are electropneumatically controlled and will close on electric or pneumatic failure. Duplication of equipment and automatic switching upon a drop in airflow protect against SBHE exhaust fan failure; should all exhaust fans fail, interlocks will shut the supply dampers and deenergize the supply fans to prevent pressurization of the building. By proper regulation of airflow, zones 3 and 4 are kept at the appropriate pressures with respect to zone 2. The only zone 3 areas that can normally be occupied are the reactor bay, beam room, and experiment room. Of these, the reactor bay is the most likely location for an activity release, and the containment philosophy is most rigidly applied here. Air locks from the bay are provided for all routinely used personnel and vehicle passages to the outside or to the water wing.

It should be noted, however, that one of the advantages of dynamic containment is that it will provide protection even if there is a large opening in the building.

Containment for individual experiments in the experiment and beam rooms will be provided as needed by small ventilated cells. Blanked-off SBHE connections are available for this purpose. Although these rooms are in zone 3, their inclusion is primarily for convenience rather than because of a real containment requirement.

Various system conditions are remotely indicated in the reactor control room. These include the pressure difference between the reactor bay and the atmosphere outside the building, the pressure difference between the beam room and outside, the difference between the experiment room and outside, and fan operating information.

4.3 Air-Conditioning Systems

The second and third floors of the control and water wing are air conditioned by a single, package type, multizone air-conditioning unit, designated AC-15. The unit is sized for approximately 18,360 cfm. It consists of a fresh-air intake louver, replaceable air filters, heating coils, chilled-water cooling coils, a slow-speed centrifugal fan, and hot and cold deck dampers. Pneumatic controls are used to automatically regulate the zone temperatures and humidity. Up to 9865 cfm may be recirculated. The areas served are maintained at a slight positive pressure with respect to the reactor bay. The conditioned air supply is conveyed throughout the area by ducts to a system of diffusers in order to provide even distribution. The AC-15 unit is located in room 101 at the northwest corner of the first floor of the water wing.

The zone 3 areas (reactor bay, beam room, and experiment room) utilize small packaged air-conditioning units requiring a minimum of distribution duct work. Equipment cells, such as those containing the demineralizers, depend upon cool air from the surrounding corridors for cooling. Each heat exchanger cell has three cooling units because of the larger heat load. The reactor bay is air conditioned by six floor-mounted individual package units. Two of these draw fresh air from outside the building through intake louvers and prefilters. The other four units recirculate air and have no fresh-air intakes, and units are equipped with individual filter chambers containing banks of prefilters and absolute filters. The filter chambers are of galvanized sheet metal with all joints sealed airtight. Access doors are provided through which the filters can be sprayed with a particle-confining coating and removed. No special shielding is required. Pressure-differential gages are mounted on the chambers to indicate the condition of the filters. These units are sized to handle a total of 38,400 cfm, of which approximately 26,400 cfm is recirculated. The remaining 12,000 cfm is removed by the SBHE system.

Air conditioning in the experiment and beam rooms is handled in a similar fashion by package units: one fresh outside air unit in each room, four recirculating units in the experiment room, and two recirculating units in the beam room. The experiment room units are sized to handle a total of 32,400 cfm, with 26,400 cfm being recirculated. The beam room units handle 19,200 cfm, with a recirculation of 13,200 cfm. The SBHE system removes approximately 6000 cfm from each of these rooms.

Chilled water at 40°F is supplied to the various air-conditioning units by a large unit which has a capacity of 4×10^6 Btu/hr. The heat collected is dumped to the secondary coolant system (see Sec. 6) through the chiller located in room 101. Low-pressure (15 psi) steam for the air-conditioning units is obtained by pressure reduction from the 125-psi steam supply. Work areas where special heating is required are supplied with unit-heaters employing heating coil—fan combinations. Semi-isolated stairwells, vestibules, offices, etc., are provided with steam-heated finned-tube radiators.

4.4 Ventilation System

Although maintained at a positive pressure with respect to the main reactor building and the shielded portions of the water wing, portions of zone 2 (including the electrical, mechanical, and process equipment rooms) are ventilated by a system which uses 100% outside air supply (see Fig. 4.4.1). Exhaust fans discharge approximately 18,500 cfm directly to the atmosphere from the water wing. The ventilation system for the electrical equipment and battery rooms G-12 and G-13 is a packaged heating and ventilating unit consisting of a fresh-air intake louver, replaceable filter, heating coil, and a slow-speed fan. Approximately 5000 cfm of air is exhausted from these rooms directly to the atmosphere. These rooms, originally intended to house lead-acid batteries (since replaced by nickel-cadmium batteries), are considered to be outside the containment.

4.5 Special Building Hot Exhaust System (SBHE)

The SBHE system is a maximum reliability system which provides secondary containment to prevent the uncontrolled release to the atmosphere of airborne activity from the primary coolant system or from fuel or other components stored in the reactor pool. It also provides secondary protection for the experiment and beam rooms.

The system has two main branches which discharge into the 250-ft stack. Each branch contains a filter system designed to remove particulate and gaseous contaminants (see Fig. 4.5.1).

One of these branches serves the reactor bay and the primary coolant heat exchanger cells. This header is embedded in the concrete building structure for shielding. The other branch exhausts air from the beam room, experiment room, pipe tunnel, and the shielded equipment rooms which house the cleanup equipment in the water wing. The arrangement of the SBHE ducts is shown in Fig. 4.5.2.

The ducts for the reactor bay system are embedded in the concrete at the periphery of the pool walls. They are provided with inlet registers immediately above the pool scuppers. Air is drawn into some of these registers from the reactor bay through an aperture in the reactor pool cover. (Fig. 5.5.1). The ducts are sized and are controlled by a manual damper to remove a minimum of 5000 cfm of air from under the pool cover, corresponding to a minimum flow velocity of 100 fpm through the 50-ft^2 pool cover aperture. The movable reactor pool cover is so arranged that during fuel handling manipulations in the reactor pool, the aperture will move with the bridge and still leave the reactor pool covered. The remainder of the peripheral duct draws approximately 6000 cfm of reactor bay air over the surface of the clean pools and the critical pool. The defective element fuel storage tanks are continuously vented into the poolside SBHE duct above pool water level. The south portion of the duct serving the clean pools may be closed with a manual damper to permit the diversion of part of the reactor bay exhaust capacity to the heat exchanger cells. In this fashion a flow of air from the bay into the cells is assured during maintenance operations which require one of the heat exchanger cell shielding plugs to be removed.

A header from the poolside SBHE branch serves to exhaust 1000 cfm directly from the northwest corner of the bay adjacent to the decontamination pad. Another header serves the four primary heat exchanger cells, with a total flow of 2300 cfm. The main reactor bay branch is run within the pool structure below the beam room, west out of the reactor building, and then north to the east section of the SBHE filter pit.

The second main branch of the SBHE system, which is not shielded, runs parallel to the first branch under the beam room floor and to the filter pit. It serves the experiment room, beam room, and the shielded equipment areas. Approximately 6000 cfm is drawn from the beam room, including 500 cfm from the subpile room. An additional 6000 cfm is exhausted from the experiment room, and 2300 cfm from the equipment cells in the water wing.

The pool peripheral exhaust duct and inlets, the exhaust from the pump and exchanger cells, the ductwork embedded in the pool structure, and all other ducts embedded in concrete are constructed of welded type 304 stainless steel. The pool concrete structure contains supplementary shielding where required to compensate for the concrete displaced by the ductwork. All this ductwork was pressure tested to assure airtightness before it was embedded in the concrete. Exposed ductwork is galvanized sheet iron in accordance with standard sheet-metal duct construction. All joints are sealed airtight.

Ductwork laid underneath soil bearing floor-slabs or buried in earth outside of the reactor building structure is constructed of extra-strong Transite pipe joined with a gastight sealing compound. The Transite pipe is coated with two-ply tar and felt to prevent water seepage into the ducts. A chevron-type baffle is installed at the filter pit to remove entrained water from the air to protect the filters.

4.6 Hot Off-Gas Systems (HOG)

The HOG systems collect and decontaminate gaseous effluent from various vessels, lines, and other equipment. They also serve local areas into which highly concentrated radioactive gases may be discharged. The decontaminated gases are then released through the SBHE filter to the 250-ft stack. Two systems are provided: the closed hot off-gas system (CHOG), connected to components from which the gases may escape under pressure; and the open hot off-gas system (OHOG), connected to unpressurized equipment (see Fig. 4.6.1). The CHOG and OHOG systems are each constructed of type 304L stainless steel, sched-5 pipe except where heavier gage pipe was required for embedment. Piping within the pool structure has additional shielding where necessary to compensate for the concrete removed to allow pipe space. Embedded pipe is pressure tested before concrete is poured to assure an airtight system.

ORNL/TM-1138/R1 gives schedule 10 (9.4.2, page 9-16)

4.6.1 Closed Hot Off-Gas System (CHOG)

The CHOG system services those components which may become pressurized, initially the primary coolant and pool deaerators. Stations designed for future connections to the CHOG system are located in the experiment room on the shield wall above each beam tube access and also high on the shield wall to permit connection to pool experiments through sleeves in the pool structure.

The CHOG ducts which originate at the collection points run to a common header and thence to the CHOG section of the filter pit. Those ducts located in occupied areas are shielded by the equivalent of 1 in. of lead. The system is sized to handle up to 500 cfm, with inlet pressures of −30 in. WG with any four of the functional connections in use.

4.6.2 Open Hot Off-Gas System (OHOG)

The OHOG system services those connections which cannot be pressurized, and also vents the intermediate-level-waste system. The following components are directly served by the OHOG system:

1. primary coolant demineralizers and filters,
2. primary coolant head tank,
3. pool coolant demineralizers and filters,
4. pool coolant filter,
5. pool surge tank.

In addition, valve stations permitting future access to the OHOG system are provided in the following locations:

1. in the experiment room on the shield wall adjacent to each engineering facility access,
2. in the experiment room high on the shield wall to permit connection to pool experiments through sleeves in the pool structure,
3. at each of the beam hole ports in the beam room,
4. in the floor of the reactor bay (two flush connections, one north and one south near the pool rim),
5. on the wall of the reactor bay adjacent to the decontamination pad.

Ductwork for the OHOG system runs separately from the CHOG duct. In occupied areas it also is shielded by the equivalent of 1 in. of lead. Like the CHOG system, the OHOG system is sized to collect 500 cfm with inlet pressures of –30 in. WG when any four of the functional connections are in use.

4.7 Filters and Exhaust Equipment

Air from the SBHE and the two HOG systems is led from the various headers to separate underground filter pits which contain both absolute and charcoal filters (see Fig. 4.7.1). After passing through the filters, the air is monitored and forced up a 250-ft stack so that any residual activity is dissipated by dispersion, diffusion, and decay. Separate filter banks and fans are provided for each of the three systems; however, the two HOG systems discharge through a common header to the suction side of the SBHE filters. Backdraft dampers on each fan prevent recirculation.

The filter pit, which includes separate sections for each system, is an outdoor, below-grade, waterproofed, concrete structure designed to withstand pressure differences between +0.5 psi and –57 in. WG. At grade level the SBHE system is shielded with 2 ft of concrete and the HOG section with 4 ft of concrete. Removable concrete plugs permit access to the filters. To facilitate decontamination, the inside of the pits is painted with a smooth protective coating which prevents radioactive material from soaking into the concrete.

4.7.1 SBHE Filters and Fans

The SBHE section of the filter pit contains three filter cells in parallel. Each branch of the system is served by a separate cell. The third (central) cell serves as a standby which can be put into service on either branch without necessitating a plant shutdown. A diagram of the system is shown in Fig. 4.7.2.

Each filter cell contains an assembly of (1) a bank of 12 glass-fiber prefilters mounted in a 3×4 array; (2) a 3×4 bank of 12 glass-fiber absolute filters; (3) a silver plated copper mesh filter; (4) two 3×4 banks of 12 charcoal absorption units, now $1\frac{1}{8}$ in. thick; and (5) a 3×4 bank of absolute filters. The filter system is designed to remove 99.97% of the particulate matter greater than 0.3 μ and to remove more than 99.99% of the elemental iodine and more than 99.5% of methyl iodide.

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 filter to the filter frames are removed remotely. Full system airflow during filter changes is maintained by diverting the flow to the standby cell.

The filter frames are of corrosion-protected carbon steel. Drainage and condensate from all low points in the exhaust ducts and the filter pit are collected and drained by gravity to the process-waste-disposal system.

Three 28,600-cfm exhaust fans driven by 75-hp squirrel-cage induction motors convey the air from the SBHE filters to the stack. These are of carbon steel construction with scrolls and wheels painted with corrosion-resistant paint. Normally only fan 3 is operated, with the other two in standby. The single duct from the SBHE section of the filter pit is split to supply the fans. Each standby fan is fed from a different generator in the normal-emergency power system to provide reliability during a normal-power outage. The automatic changeover from the operating fan to a standby fan is actuated by airflow switches which detect malfunction of the operating unit. Once a fan is started, it can only be shut down manually. Backflow through a standby fan is prevented by a gravity damper.

Pitot flow elements are installed in the SBHE ducts upstream of the filter pits to monitor flows. These flow signals are transmitted to, and displayed in, the control room. On a significant loss of flow, the standby fans are automatically started; on a further loss of flow, the air-conditioning intake fans are stopped. The fan outlet duct and the stack are monitored for radioactive material which has escaped the filters; however, high stack activity will not cause automatic shutdown of the system fans.

4.7.2 HOG Fans and Filters

The filters for the CHOG and OHOG systems are located in a section of the filter pit separated from the SBHE filters by a concrete wall. Three filter cells (see Fig. 4.7.3) are used: one for the CHOG system, one for the OHOG system, and a common standby for use during filter changes. The filter arrangement in each cell consists of a glass-fiber prefilter, an absolute filter, a silver-plated copper-wool filter, two charcoal absorbers in series, and a final absolute filter (now identical to SBHE filters except for overall dimensions). All exposed metal in the cell is stainless steel.

Changeover to the standby cell is accomplished by means of externally operated manual dampers. The filters may be changed, without interrupting plant operation, in a manner similar to that described for the SBHE system. Air from the HOG systems enters the filter pit through two ducts, one for each system. These may be suitably dampered to permit full system airflow during a filter change.

Four identical 500-cfm exhaust fans driven by 7 1/2-hp squirrel-cage induction motors are provided for off-gas service. Each system is equipped with two fans, one of which is normally operating while the other is in standby. The outlet ducts from the filters are split to permit automatic fan changeover. One fan from each system is connected to normal-emergency power system No. 1, and the other two are connected to normal-emergency power system No. 2. Pressure controllers are used to detect malfunction of the normally operating fan and to start the standby fan. The fan changeover is actuated automatically; but, once started, a fan must be manually shut down. Gravity dampers prevent backflow through the standby unit. Various system parameters are displayed in the control room.

The common exhaust header for the HOG fans is returned to the inlet of the SBHE filter pits to provide an added measure of decontamination.

4.8 Gaseous Waste Stack

The HFIR stack is a free-standing, reinforced concrete stack lined with acid-resistant brick. The connecting breeching to the exhaust fans is constructed of carbon steel plate painted with corrosion-resistant paint. The stack is 250 ft high with its discharge at elevation 1085 ft. The inside diameter at the top of the stack is 5 ft; the inside diameter at the base is 10 ft 9 in.

The stack provides for 30,000-cfm exhaust capacity and has an additional capacity of 30,000 cfm to handle TRU requirements. It was designed to discharge 60,000 cfm at 130°F. Under design conditions the discharge velocity is ~3050 fpm. With only the ~30,000-cfm HFIR load, the discharge velocity is 1525 fpm. Provisions are made for sampling gas and measuring flow in the main ducts and in the stack itself. Installed radioactivity monitors continuously indicate the amount of activity discharged to the atmosphere. Detectors include β-γ and α particulate monitors, an inert-gas monitor, and an iodine monitor on the stack and a β-γ particulate monitor in the HFIR stack breeching.

An open-shed-type structure near the base of the stack provides weather protection for the fans, and a small enclosure in this shed houses the instrumentation.

4.9 Exhaust Systems Instrumentation and Control

It is necessary that these exhaust systems retain a high degree of reliability even under abnormal conditions. Therefore, considerable effort has been expended to provide adequate instrumentation and control and to ensure the availability of power to drive the fan motors. The normal-emergency electrical supplies are described in some detail in Sec. 10.1; and, in the following descriptions, some familiarity with these systems is assumed. Additional information concerning radioactivity monitoring may be found in Sec. 8.7.

4.9.1 SBHE Instrumentation and Control

Each of the SBHE standby exhaust fans (FN 1 and 2) is provided with an "on-off-standby" selector switch located at the fans. In normal operation fan SBHE-3 operates continuously and is provided only with a local "on-off" switch. Indicating lights on the panel board, operated by motor contacts and the standby switch, provide information concerning the condition of the fans.

Twelve airflow monitoring switches, six in each of the main branches of the SBHE system upstream of the filter pit, provide two-stage alarm and control action (see Fig. 4.9.1). A drop in airflow to $<85\%$ of normal in either branch will cause a flow switch in that branch to open and actuate a visual and audible alarm in the control room. A drop in airflow to $<80\%$ of normal in either duct will cause one or more of four switches, two in each branch, to open. This condition starts a standby fan and sounds an alarm in the control room. A further decrease of flow to $<75\%$ of normal will cause one or more of four other switches, two in each branch, to open. This condition shuts down the air-conditioning units which draw fresh air into the zone 3 containment areas. When normal flow is restored, the air-conditioning units will restart automatically. These control circuits are designed so that a loss in control power or the opening of a relay coil will simulate a loss of airflow. A gravity relief damper (BDD) in the fresh-air intake unit will start to open at a pressure of -0.4 in. WG to provide protection against excessive negative pressures. No coincidence circuits are used in these flow switches; any one of the switches can cause an alarm, a standby fan startup, or an air-conditioning unit shutdown. The operating conditions and functions of these pressure switches are listed in Table 4.9.1.

The SBHE fans FN-1 and FN-2 are connected to normal-emergency power systems No. 2 and No. 1 respectively. The circuits are arranged so that the stack fans receive high priority on both emergency diesel generators. SBHE fan SBHE-3 is connected to normal power.

A continuous display of the differential pressure between the zone 3 containment areas and the outside atmosphere is given at the gaseous-waste-system panel board in the control room. This is obtained from three pressure gages (Table 4.9.2) which monitor the experiment room, beam room, and the reactor bay. The nominal differential pressure is -0.3 in. WG. Because neither the beam room nor the experiment room are provided with air locks, it is expected that the differential pressure in these rooms will fluctuate to a greater extent than that in the reactor bay. Modulating dampers are provided in the fresh-air intakes to minimize the fluctuations. The reactor bay is provided with air-lock entrances for both personnel and motorized equipment. These are not true air locks, such as those found in "static containment" systems, but are small antechambers which serve a similar purpose. A visual and audible alarm is given in the control room should the differential pressure in the reactor bay fall below -0.10 in. WG.

Four pitot tubes, two in each branch, are provided to monitor the flow through the SBHE system; they also provide the airflow signals which operate the twelve flow-monitoring switches. The flow indicators are located on the gaseous-waste-system panel in the control room.

Table 4.9.1. SBHE Pressure Switch Data

Switch No.	Duct Location	Normal Pressure[a]	Setpoint Pressure[a]	Type of Action
PS-914B and 904B	East	0.55	0.375 (85% flow)	Alarm
PS-913B and 903B	West	0.55	0.375 (85% flow)	Alarm
PS-914A1 and 904A1	East	0.55	0.30 (80% flow)	Starts standby fan and alarms
PS-913A1 and 903A1	West	0.55	0.30 (80% flow)	Starts standby fan and alarms
PS-914A2 and 904A2	East	0.55	0.275 (75% flow)	Shuts down AC units 2, 5, 10, and 14
PS-913A2 and 903A2	West	0.55	0.275 (75% flow)	Shuts down AC units 2, 5, 10, and 14

[a]Indicates velocity head in inches of H_2O.

Table 4.9.2. Differential Pressure Gage Data

Gage No.	Area Monitored	Normal Pressure Reading (in. WG)	Alarm Setpoint (in. WG)
PdI-900	Reactor bay	0.30	0.10
PdI-901	Beam room	0.30	None
PdI-902	Experiment room	0.30	None

4.9.2 HOG Instrumentation and Control

The instrumentation for the HOG systems (Fig. 4.9.2) is quite similar to that utilized by the SBHE; however, static vacuum is monitored instead of flow. The fan controls are similar to those in the SBHE. The CHOG instruments and controls are identical to the OHOG instruments and controls except that the setpoints are different.

The CHOG fans FN-3 and FN-4 are each provided with an "on-off standby" switch at the fans. Normally one fan is running while the other is on standby. Indicating lights on the panel board indicate the condition of each fan.

Two electropneumatic pressure sensors in the CHOG duct, which transmit a signal for display in the control room, provide alarm and standby fan startup signals. A drop in the CHOG vacuum on the inlet side of the filters (from −48 to −43 in. WG) will cause an alarm in the control room. The loss of vacuum may be due to fan failure or filter plugging. Should it be due to fan failure, a pressure switch located in the fan inlet will be affected by the loss of vacuum and will actuate a relay, starting the standby fan. An alarm is also given in the control room. The fan vacuum is displayed at the fan house, and the fan running lights will indicate in the control room which of the two conditions has occurred. A similar pair of switches in the OHOG system controls fans FN-5 and FN-6; only the setpoints may differ. The pressure switches in the HOG system are listed in Table 4.9.3.

Table 4.9.3. HOG Pressure Switch Data

Switch	Duct	Vacuum (in. WG)		Action
		Normal	Setpoint	
PS-905-1	OHOG (inlet to filter)	−48	−43	Alarm
PS-915-1	OHOG (inlet to fan)	−54	−41	Alarm, standby fan starts
PS-906-1	CHOG (inlet to filter)	−48	−43	Alarm
PS-916-1	CHOG (inlet to fan)	−54	−41	Alarm, standby fan starts

No other systems are affected by these low vacuum alarm and fan startup signals. Just as in the SBHE system, the switches and relays are designed to "fail safe" on loss of control power or relay failure.

The CHOG fan FN-3 and OHOG fan FN-5 are connected to normal-emergency power system No. 2; CHOG fan FN-4 and OHOG fan FN-6 are connected to normal-emergency power system No. 1. As in the case of the SBHE fans, the HOG fans receive high priority on both emergency diesel generators.

5. REACTOR AND EXPERIMENTAL FACILITIES

5.1 Introduction

The prime purpose of the HFIR is to establish, within a limited volume, an unperturbed thermal-neutron flux of 5×10^{15} neutrons cm^{-2} sec^{-1} for producing milligram quantities of Cf^{252} annually. Analysis of the transplutonium production schemes[1] and design studies[2,3] led to the conclusion that this could be accomplished, within the scope of existing technology, with a light-water flux-trap type of reactor fueled with uranium highly enriched in U^{235} and reflected by beryllium. The reactor power level was established at 100 Mw by a combination of economic considerations and flux requirements. The size and configuration of the core were determined by nuclear and heat-removal optimization studies. These, together with thermal and hydraulic considerations, determine the power density to be achieved.

The reactor core, shown in Fig. 5.1.1, consists of four concentric regions, each approximately 2 ft high. The central region, often called the "island," has the highest thermal-neutron flux and constitutes the flux trap. The island is approximately 5 in. in diameter and will contain the Pu^{242} target, which is in the form of a bundle of rods.

The fuel region, located immediately outside the island, has an inside diameter of approximately 5 in. and an outside diameter of approximately $17\frac{1}{8}$ in. It consists of two cylindrical fuel elements containing initially 9.4 kg of U^{235} contained in vertically oriented curved plates. To minimize the radial peak-to-average power density ratio, the fuel concentration is radially graded. To further flatten the power distribution and aid in reactor control, the inner element typically contains 2.8 g of B^{10}.

The control region, with control plates in the form of two thin poison-bearing concentric cylinders, is located between the outer fuel element and the beryllium reflector. Each cylinder contains an axially varied poison to minimize the axial value of the peak-to-average power density ratio throughout the core lifetime.

The reflector region is a concentric ring of beryllium reflector approximately 1 ft thick. This, in turn, is reflected by water of effectively infinite thickness. Water above and below the core serves as an end reflector.

To increase the usefulness of the reactor, certain experimental facilities have been provided in addition to the flux trap. These include two horizontal beam tubes, one extending radially and the other tangentially from within the beryllium reflector; a horizontal tangential beam tube which passes completely through the reflector; four slant access facilities to the outside edge of the reflector; and 38 vertical holes in the reflector. Various thimbles and access facilities for nuclear instrumentation penetrate the reactor vessel.

The reactor core is contained in an 8-ft-diam pressure vessel. It is cooled by demineralized water which, under normal conditions, is pumped through the system at the rate of about 16,000

---

[1] J. A. Lane *et al., High Flux Isotope Reactor Preliminary Design Study,* ORNL-CF-59-2-65 (Mar. 20, 1959).

[2] R. D. Cheverton, *HFIR Preliminary Physics Report,* ORNL-3006 (Oct. 4, 1960).

[3] N. Hilvety and T. G. Chapman, "Thermal Design of the HFIR Fuel Element," in *Research Reactor Fuel Element Conference, September 17–19, 1962, Gatlinburg, Tennessee,* TID-7642, Book 1.

gpm. This corresponds to a velocity of 51 fps through the fuel region and 40 fps through the target array. The pressure vessel is located in a pool of water (Fig. 3.1.1) 18 ft in diameter and 36 ft deep. This water serves as a biological shield and also provides access to the reactor. The reactor pool is connected to a rectangular storage pool 41½ ft long, 20 ft deep, and 18 ft wide. The general arrangement of the reactor components is shown in Figs. 5.1.2 and 5.1.3, with a detailed assembly shown in Fig. 5.1.4.

5.2 Core Components

The reactor core components, which include the target array, fuel elements, control plates, and beryllium reflector, are contained in the reactor vessel. They are supported on two concentric cylinders (Fig. 5.3.1), called pedestals, which are bolted to a cylindrical member called the fuel and reflector support sleeve assembly. This is in turn bolted to the lower part of the pressure vessel. A pair of cylindrical shrouds extend above the top surface of the fuel element. These shrouds enclose the upper part of the control cylinders. This provides the required coolant flow characteristics and protects the control cylinders from the turbulence of the vessel inlet coolant stream.

5.2.1 Target Assembly

The target assembly, shown in Fig. 5.2.1, is located in the island and contains the Pu^{242} target material. It consists of 30 rods (⅜ in. OD) spaced in a triangular pattern. The rods contain pellets fabricated from a mixture of PuO_2 and aluminum powder. The pellets are then packed in aluminum tubes and capped, and the tubes are collapsed onto the pellets to provide good heat transfer. Appropriate leak tests are applied to each completed tube. The target tubes are centered inside $^{21}/_{32}$-in. diameter circular shrouds, thereby providing each rod with its own cooling channel. The shrouds change to a ⅝-in. hexagonal shape near the upper end in order to provide for spacing and clamping. A typical target rod is shown in Fig. 5.2.2. The target holder serves to securely hold the array of target rods, to accurately position the assembly, and to facilitate removal and maintenance operations. The hydraulic tube is described in Sec. 5.6.3.

Optimization studies[2] indicate that an initial charge of 310 g of $Pu^{242}O_2$ is desirable. This will result in a maximum heat-generation rate in the target of approximately 900 kw. The array of rods provides adequate heat transfer characteristics when positioned on a ⅝-in. triangular pitch.

A design water flow rate of approximately 788 gpm is maintained through the island region of which 675 gpm flows around the target rods. This flow corresponds to a coolant velocity of about 40 fps in the target region and is more than sufficient for adequate heat removal. The average heat flux at the target rod surface is calculated[4] to be about 0.6×10^6 Btu ft^{-2} hr^{-1}. This average occurs during heat-generation peaks,[5] as illustrated in Fig. 5.2.3. With a hot spot heat flux of 1×10^6 Btu ft^{-2} hr^{-1} and a 0.1-mil gas gap between the pellet and the rod wall, the internal temperature is not expected to exceed 900°F at the 900-kw heat-generation rate.

Calculations indicate that 215 STP cm^3 of gas containing 91.6% Xe, 6% He, and 2.4% Kr will be produced during an 18-month irradiation of a target rod initially containing 10 g of $Pu^{242}O_2$ with 1% Pu^{239} and 1% Pu^{241}. Approximately half this volume would be produced in subsequent curium recycle rods. The rods are designed with a 6.5-cm^3 void to accommodate the gaseous fission products. The target tubes have sufficient strength to resist the 1000 psi external coolant pressure or the internal pressure resulting from 100% fission gas evolution.

Corrosion studies indicate that the corrosion rate of the X-8001 aluminum will not, under the imposed conditions, exceed 0.005 in./yr.

---

[4] T. G. Chapman, *HFIR Target Design Study*, ORNL-TM-1084 (Sept. 3, 1965).

[5] H. C. Claiborne and M. P. Lietzke, *Californium Production in the High Flux Isotope Reactor*, ORNL-CF-59-8-125 (August 1959).

5.2.2 Fuel Elements

The fuel region of the HFIR, shown in Fig. 5.2.4, is composed of two concentric, cylindrical fuel elements containing vertically oriented curved plates extending in the radial direction. The inner element, which contains 171 plates, is initially loaded with 2.6 kg of U^{235} and 2.8 g of B^{10}. The inner diameter is 5.067 in., and the outer diameter is 10.590 in. The outer element contains 369 plates and initially contains 6.8 kg of U^{235} but no burnable poison. Its inner diameter is 11.250 in., and the outer diameter is 17.134 in.

The individual plates are of a sandwich-type construction composed of a fuel-bearing cermet hermetically sealed between covers of type 6061 aluminum. The fuel-bearing cermet is a mixture of U_3O_8 and aluminum, approximately 30 wt % U_3O_8 in the case of the inner annulus and 41 wt % U_3O_8 in the case of the outer annulus. Uranium containing at least 93% U^{235} is used as fuel. Initially each inner annulus plate contains 15.18 g ± 1.0% of U^{235}; each outer annulus plate contains 18.44 g ± 1% of U^{235}. In addition, each inner annulus plate initially contains 0.0164 g of B^{10}. The finished plates are 0.050 in. thick and 2 ft long, with a nominal active length of 20 in. The thickness of the fuel plate core is 0.030 in., and the cladding thickness is 0.010 in. The plates are curved in the form of an involute which provides a constant cooling channel width between plates; the width is 0.050 in. Before bending, the inner annulus plates are approximately 3.6 in. wide, and the outer annulus plates are approximately 3.2 in. wide. The fuel cores are centered to provide adequate edge cladding. Both inner and outer plates are fastened between cylindrical aluminum side plates by welding.

To minimize the radial peak-to-average power density ratio, the fuel surface density in each plate is varied along the arc of the involute curve. The B^{10} included in the inner plates in the form of B_4C, is added to an aluminum filler piece, which together with the fuel-bearing cermet makes up a composite, rectangular fuel plate core, as shown schematically in Figs. 5.2.5(*a*) and (*b*).

A fuel element has a typical lifetime at 100 Mw of about 23 days. Under normal operating conditions, flow through the fuel is ~13,000 gpm, which corresponds to a velocity of 51 fps. The core pressure drop at this velocity is ~110 psi. At a total heat power of 100 Mw, the average heat flux is 0.78×10^6 Btu ft^{-2} hr^{-1}, and the calculated maximum hot spot heat flux is 1.97×10^6 Btu ft^{-2} hr^{-1}. The minimum incipient boiling power level under anticipated conditions – inlet pressure, 600 psi; flow, 51 fps; inlet temperature, 120°F – is calculated to be 142 Mw at the beginning of the operating cycle. The pertinent thermal characteristics of the core at design conditions are given in Appendix A.

5.2.3 Beryllium Reflector

The fuel region is radially reflected by a 1-ft-thick beryllium reflector. The beryllium reflector, in which are located the experiment facilities, is shown in Figs. 5.1.1, 5.2.6, and 5.2.7. The inner portion of the reflector is readily removable to permit replacement, when necessary due to radiation damage or other causes, and to allow access to the control plate drives. The reflector is water cooled; but in order to avoid excess neutron absorption, the volume of water in the reflector is held to the minimum consistent with the cooling requirements.

The inner portion of the reflector consists of a 3.689-in.-thick beryllium section composed of four concentric cylinders with cooling water flowing axially between them. The outer part of the reflector is an 8.375-in.-thick beryllium annulus provided with axial circular coolant channels. Three of the four inner cylinders, called the removable reflector, may be removed from the reactor as a unit, thus permitting access to the control plate shock absorbers. The fourth cylinder, called the semipermanent reflector, is removed only when necessary because of radiation damage. Four small pieces of the semipermanent beryllium, called control plate access plugs, are easily removable to facilitate access to the control drives. Although the thick outer annulus, called the permanent reflector, may be removed, it is anticipated that this would be done very infrequently.

To minimize the chance, should the beryllium crack, of interference with the control plates which are located between the removable reflector and the fuel region, a $\frac{1}{16}$-in.-thick aluminum liner covers the inner surface of the removable beryllium. Coolant for these removable reflector cylinders is provided by water flowing through $\frac{1}{8}$-in. axial grooves on the surfaces of these cylinders. A $\frac{1}{16}$-in.-thick water gap between the removable and semipermanent reflectors facilitates removal and replacement of the removable reflector, furnishes cooling, and provides for expansion of the removable reflector. Eight $\frac{1}{2}$-in.-diam vertical irradiation facilities are provided in the middle removable reflector cylinder, and eight others are located in the removable control plate access plugs. They are provided with beryllium plugs which will be in place when the facilities are not in use.

The permanent reflector is cooled by water which flows axially through $\frac{1}{8}$-in.-diam holes which are spaced on circles concentric with the reflector. There are five such circles, each containing 80 cooling holes, except where the coolant hole pattern is interrupted by the experimental facilities.

The permanent reflector is penetrated by 16 vertical 1.58-in.-diam and 6 vertical 2.83-in.-diam experimental facilities similar to those in the removable reflector. Two 4-in. beam tubes penetrate the permanent reflector and terminate in the semipermanent beryllium. One 4-in.-diam tangential beam tube penetrates all the way through the reflector assembly. Four slanted grooves in the outer surface of the reflector accommodate the engineering test facilities.

5.2.4 Control Plates

The control plates are located in a 0.869-in.-thick annular region between the outer fuel element and the removable beryllium. The control plates consist of two $\frac{1}{4}$-in.-thick concentric cylinders. The coolant passages, listed radially outward, are 0.104-in., 0.170-in., and 0.095-in. The coolant water velocity in the control region is about 16 fps. The inner cylinder, which is a single piece, is used both as a shim and a regulating rod. The outer cylinder is divided into four quadrants, each used as a shim-safety rod and each having its own drive rod and scram mechanism (see Secs. 8.1 and 8.6 for complete description of control action). The general arrangement is shown in Figs. 5.1.1 and 5.2.8.

The chain reaction is controlled by altering the efficiency of the beryllium reflector as the control plates are moved vertically between the core and the reflector. The single inner cylinder is driven downward, thus driving the poison out of the reactor to increase reactivity, while the outer quadrants are driven upward out of the reactor to increase reactivity.

In order to reduce axial variations in the power distribution and to prolong the life of the control elements the cylinders are divided into three longitudinal sections, each of which has different neutron absorbing characteristics. The lower section of the inner cylinder and the upper section of the outer cylinder contain 33 vol % Eu_2O_3 dispersed in aluminum. These sections are highly neutron absorbing and are called the black regions. The central section of each plate contains 40 vol % tantalum dispersed in aluminum. They are less absorbing than the Eu_2O_3 sections and are called the gray regions. The upper section of the inner cylinder and the lower section of the outer cylinder are made of aluminum and are called the white regions. In normal operation criticality is achieved by driving in unison the inner cylinder down and the outer quadrants up. As shown in Fig. 5.2.9, the two control cylinders are then gradually withdrawn to compensate for fuel burnup throughout the core life and are maintained symmetrical about the core midplane so as to maintain symmetry of the power density in the axial direction. A servo control system maintains constant power by moving the inner cylinder.

In each control element the black region is 22 in. long, $\frac{3}{16}$-in.-thick and is clad with $\frac{1}{32}$-in.-thick aluminum. The gray regions are 5 in. long and have the same thickness of absorber and cladding as is the case in the black region. The white regions are of solid aluminum. In order to balance the hydraulic forces, a large number of $\frac{1}{4}$-in. holes are drilled through the white and the gray regions of the five control elements. Corrosion studies indicate that edge cladding around these holes is not necessary in the gray region.

The inner control cylinder has an outer radius of 8.921 in. and an inner radius of 8.671 in. The overall length is $68\frac{3}{8}$ in. The lower end of the black region is $14\frac{5}{16}$ in. above the lower edge of the cylinder, and the upper end of the white region is $7\frac{1}{16}$ in. below the upper edge of the cylinder. The outer control plates have an outer radius of 9.300 in. and an inner radius of 9.050 in. The overall width of each is $13\frac{39}{64}$ in., and the overall length is $66\frac{3}{16}$ in. The upper end of the black region is $8\frac{5}{8}$ in. below the upper end of the plate, and the lower end of the white region is $10\frac{9}{16}$ in. above the lower end of the plates.

The cylinder and plates are driven from beneath the reactor by drive rods which extend into the subpile room, where the drive mechanisms are located. These mechanisms are described in detail in Sec. 8. Guidance for each outer control plate is furnished by six ball bearings. Four bearings are attached to each outer control plate, two at each end, and are run in stationary tracks which extend above and below the core. The other two bearings are attached to the lower track assembly and bear against the control plate. The inner control cylinder is guided by eight bearings, four of which are attached to the track assemblies at each end of the reflector and extend through the slots between the four outer plates.

5.3 Core Support and Assembly

The reactor core is supported by pedestals which are fastened to the bottom of the pressure vessel. An exploded view of the various components showing their arrangement is shown in Fig. 5.3.1. The main support member is the stainless steel fuel and reflector support and sleeve assembly which rests on the core support ring and extends into the lower tank extension (see Fig. 5.4.1). It is held in place by bolts and provides support for three pedestals, which in turn support and guide the core components and track assemblies.

The reflector container and support pedestal is a 6061 aluminum hollow cylinder approximately $3\frac{1}{2}$ ft in diameter by $2\frac{1}{2}$ ft high. It is bolted to the edge of the support assembly. The reflector rests on top of this cylinder and is clamped between the pedestal and outer shroud by external tie plates. A second pedestal, the fuel grid support pedestal, is also in the form of a hollow cylinder. It is located inside, and concentric with, the reflector pedestal; it is bolted to the center of the fuel and reflector support assembly and is slotted to accommodate the inner control plate drive. This pedestal is approximately 1 ft 3 in. in diameter and 3 ft high. The stainless steel fuel grid provides mounting surfaces for the fuel elements. It rests on the fuel grid support pedestal.

The control element guidance system consists of two track assemblies; one above and one below the core region. The lower track assembly is supported by the fuel and reflector support and is aligned by the track pedestal. The upper track assembly is supported and aligned by the outer shroud.

Two concentric shrouds are located above the core to provide proper distribution of coolant flow, to protect the control cylinders from the high-velocity water, and to furnish support and alignment features for various components above the core. The outer shroud is clamped to the reflector support pedestal. A 3-ft-long cylindrical extension surrounds the control region. The lower portion serves as a flow-distributing plenum for the reflector. The inner shroud is also cylindrical and fits over the outside edge of the outer fuel element. It extends upward inside the control plate region. A 304L stainless steel flange closes the top between the two shrouds. The inner shroud is easily removable and must be lifted out to change fuel. To provide access to the control plates and the removable beryllium, the shroud flange must also be removed. The outer shroud may be removed but remains in place during all routine core manipulations. Suitable orifices in the shroud flange and between the shroud flange and inner shroud ensure adequate cooling to the reflector and control plate regions. The outer shroud is fabricated of 6061 aluminum and the inner shroud of type 347 stainless steel.

The components are designed to withstand the applicable combination of loads, including a 125-psi pressure drop, hydraulic loads, dead weight, and any extra loads resulting from control drive malfunction. Mechanical and thermal stresses have been analyzed and found in all cases to be well within the elastic limits.

To minimize the probability of blocking the coolant channels in the reactor core, a strainer is located in the primary coolant loop at the point where the loop branches to form the two inlet lines to the reactor vessel. The strainer, a multilayered stainless steel wire cloth, is designed to retain foreign objects larger than 0.023 in. in diameter. It is located in a special spool piece in the primary coolant supply line within the reactor pool. Vertical access to the strainer basket is provided by means of a quick-opening hatch. The strainer is located sufficiently far below the pool surface to provide adequate shielding.

5.4 Reactor Pressure Vessel

The HFIR core is contained in a 94-in.-ID pressure vessel constructed of carbon steel which is clad on both sides with austenitic stainless steel. The general arrangement is shown in Fig. 5.4.1. A stainless steel extension is attached to the lower end of the vessel to permit drive rod access to the core through the 7-ft-thick subpile room ceiling. The carbon steel vessel wall is $2\frac{7}{8}$ in. thick. A minimum cladding thickness of $\frac{1}{8}$ in. is provided on the inside, with a minimum of 0.1 in. on the outside. The materials and construction are described in Appendix A.

The flat top head of the vessel is made of 14-in.-thick carbon steel clad with stainless steel (Fig. 5.4.2). It is secured to the vessel with bolts and sealed with an elastomer O-ring. A 30-in.-diam quick-opening access hatch has been provided in the head for ease of access in routine operation. It is designed with a breech lock type of closure and sealed with an elastomer O-ring piston seal. In the center of the quick-opening hatch is a 12-in.-diam target access plug with a sheer-block type closure and an elastomer O-ring piston seal.

The bottom head for the control plate drive rod access extension is a $6\frac{1}{2}$-in.-thick flat bolted head with an elastomer O-ring seal (Fig. 5.4.3). All other closures are standard ASA bolted flanged joints with metallic ring joint seals.

Aside from the quick-opening hatch and the three water inlet and exit lines, the vessel is provided with 56 penetrations (Table 5.4.1).

5.4.1 Vessel Design and Inspection

Materials have been specified in accordance with the ASME Pressure Vessel Code. To eliminate corrosion caused by normal operation and by the action of decontamination solutions, stainless steel has been utilized for surfaces exposed to the primary and pool water.

The vessel has been designed in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code and Nuclear Code Cases 1270N, 1271N, and 1273N. The operating, design, and test conditions for the vessel are:

	Operating	Design	Test
Temperature (°F)	120–167	200	~75
Pressure (psig)	650	1000	1550

Structural analysis has been performed by the methods described in the Department of Commerce Bulletin, PB 151987, "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components," December 1958 revision. The loading conditions evaluated and the results of the analysis are given in the Allis-Chalmers Design Report for the vessel.

The vessel was inspected in accordance with the requirements of the ASME code and Nuclear Code Case 1273N. Additional inspections performed include dye-penetrant and, as applicable, magnetic-particle inspection of all welding, including overlay cladding, both before and after stress relief. Additional tests included impact testing of ferritic materials and ultrasonic testing of plates, forgings, and other vessel parts. To verify the adequacy of the vessel, hydrostatic tests at 1550 psig and supplementary tests at 1000 psig following 17 cycles from atmospheric pressure to 1000 psig were performed. Measurements were made of the top head bolt tensile stress under preload and at 1550 and 1000 psig.

Table 5.4.1. Reactor Pressure Vessel Penetrations

Nomenclature	Number	Size
Cooling water inlet	2	16-in.-OD pipe
Cooling water exit	1	18-in.-OD pipe
Slant engineering facility penetration	4	6 in. 600 lb RJ flange
Ion chamber penetration	3	4 in. 600 lb RJ flange
Rabbit-hole penetration	8	3 in. 600 lb RJ flange
Radial horizontal beam tube penetration	1	14 in. 600 lb RJ flange
Tangential horizontal beam tube penetration	3	8 in. 600 lb RJ flange
Vertical experimental penetration[a]	12	2 in. 600 lb RJ flange
Vent penetration	1	4 in. 600 lb RJ flange
Vertical experimental penetration[a]	9	4 in. 600 lb RJ flange
Bottom head	1	30-in. flange
N^{16} and quadrant thermocouple[b]	4	1.080 in. diam
Fission chamber[b]	3	1.080 in. diam
Drive rod shafts[b]	5	1.938 in. diam
Top head	1	103 in. OD
Quick-opening hatch[a]	1	30 in. diam
Target access plug[c]	1	12 in. diam
	60	

[a]Located in top head.
[b]Located in bottom head.
[c]Located in quick-opening hatch.

The sizing of nozzle openings and the arrangement of the water annulus outside the core have been fixed to limit the fast-neutron dose to the pressure vessel. On this basis, vessel materials were used which assure that the nil ductility temperature, as determined by Charpy V-notch impact tests, of any portion of the vessel will never exceed +10°F after 20 yr. The operating procedures require that the vessel temperature be maintained above 70°F whenever the stress level is more than 5000 psi.

As a further precaution, radiation damage is monitored by the insertion of a number of actual pressure vessel steel samples at locations corresponding to the areas of highest exposure. Specimens are removed periodically and tested for strength and notch toughness.

5.5 Pools

The reactor vessel and the spent fuel storage facilities are located in a system of water-filled pools which furnish biological shielding, and, in the case of the spent fuel elements, also provide a mechanism for removing heat generated by the decaying fission products. The HFIR pool consists of four separate compartments (see Fig. 3.1.1):

1. A 36-ft-deep reactor pool, the lower portion of which is an 18-ft-diam by 16-ft-deep cylinder. The upper portion is an 18-ft-wide by 20-ft-long rectangle.
2. Two 20-ft-deep clean pools, each of which is 18 ft wide by $20\frac{3}{4}$ ft long.
3. A 25-ft-deep by 8-ft-diam critical pool for future use.

The reactor pool contains the reactor vessel, two defective fuel element storage tanks, and two normal spent fuel storage racks. The clean pools contain the long-term spent fuel storage facilities (see Sec. 11.6). The critical pool is designed to accommodate a future HFIR critical facility.

The pool walls and floor are constructed of reinforced concrete and are lined with type 304 stainless steel. The lining on the sides of the clean pools, the critical pool, and the rectangular portion of the reactor pools is $\frac{1}{4}$ in. thick. The lining on the pool floors and the walls of the cylindrical section of the reactor pool is $\frac{3}{8}$ in. thick. The reactor and clean pools are separated by removable aluminum gates which are provided with hatches near the bottom to facilitate the transfer of spent fuel elements between pools. A concrete wall, with a transfer hatch, separates the clean pool and the critical facility pool.

The walls of the cylindrical section of the reactor pool (Fig. 3.2.5) are of 12-ft-thick ordinary concrete. Four beam tube access holes penetrate the reactor pool wall at elevation 820 ft 6 in.; and in these regions the wall is of high-density concrete $7\frac{1}{2}$ ft thick. Additional movable shielding is provided, as required, around these openings. The walls of the rectangular section of the pool (that section above elevation 833 ft) are of 3-ft-thick ordinary concrete. Concrete block, supported by the 12-ft-thick portion of the wall, may be stacked against these walls as required to shield the vertical access facilities to the beam tubes and the slant access penetrations to the pool. The floor of the reactor pool, which is also the ceiling of the subpile room, is of high-density concrete and is 7 ft thick.

The south wall of the clean pool is constructed of 5-ft-thick ordinary concrete, and the north wall, which also serves as the south wall of the heat exchanger cells, is 4-ft-thick ordinary concrete. The walls of the critical pool are also of ordinary concrete a minimum of 6 ft thick, with barytes concrete at the level of a future core. Except for the stairwell leading to the subpile room, the area under the clean pool and critical pool is unexcavated. The floors of the pools are of ordinary concrete; the floor of the clean pool is 3 ft thick and the floor of the critical pool is 2 ft thick. The pool floor above the stairwell to the subpile room is of high-density concrete (1 ft 10 in. thick) backed up by 14 in. of steel armor plate.

Three 3-in.-diam pipe sleeves through the north wall of the clean pool at elevation 837 ft furnish access to three of the heat exchanger cells. In addition to the experiment facilities (see Sec. 5.6), six 6-in.-diam pipe sleeves at elevation 841 ft permit access to the reactor pool from the experiment room. Four similar sleeves at the same elevation provide access to the clean pools. When not in use these sleeves are closed with blank flanges and are shielded with bags of shot.

The pool liner is fabricated of stainless steel sheets welded together. The wall liner is fastened to the concrete by stainless steel rods which are welded to the underside of the liner and which penetrate the concrete walls. The floor liner is plug welded to steel bars set in the concrete floor. These attachments serve the double purpose of holding the liner in place and of providing conducting paths for the removal of heat generated by gamma-ray absorption in the concrete.

Stainless steel scuppers are provided along the inner edges of the pool at elevation 848 ft, which is 3 ft 2 in. below the reactor bay floor. These scuppers, which are 10 in. wide and 19 in. deep, convey the overflow from the pools to the pool cooling system. This small, constant flow prevents dust from collecting on the pool surface.

5.5.1 Pool Cover and Personnel Bridge

The personnel bridge spans the pool from north to south (see Fig. 5.5.1). It is a 6-ft-wide 20-ft-long platform covered with vinyl asbestos tile and surrounded by a suitable railing. The bridge is mounted on wheels and is driven on a track in the east-west direction by a 1-hp motor and gear train at a speed of 10 fpm. The tracks extend the entire length of the pool, thus permitting the bridge to be positioned over any point in them. Although the bridge drive is designed to stop the movement within 2 to 4 in. after the power is switched off, a manual drive is provided to facilitate accurate positioning. A small manually driven carriage mounted on tracks on the bridge permits accurate positioning of tools or equipment in the north-south direction.

The reactor pool is covered by $\frac{1}{2}$-in.-thick Plexiglass panels supported by aluminum beams to provide additional containment and protection (see Fig. 5.5.1). There are two covers, each 19 ft 7 in. wide and 10 ft $4\frac{1}{2}$ in. long, separated by an aperture which extends across the pool under the personnel bridge. Air from the reactor bay is drawn through the aperture and into the special building hot exhaust (SBHE) system ducts (see Sec. 4) which are located just above the scuppers in the pool walls. In this way any vapor or gas originating in the pool is prevented from escaping into the reactor bay.

The aperture is nominally 3 ft wide but is adjustable to provide adequate air velocity. The cover itself is carried on a wheel-and-track arrangement. Two long pins operated from the bridge, one on the east and one on the west side, engage it so that it will follow the bridge as it is moved during fuel handling. The reactor bay floor at the west end of the reactor pool is a steel plate supported on beams with sufficient room underneath to accommodate the west end of the cover, thus permitting the aperture to traverse the entire reactor pool.

5.6 Experimental Facilities

Although the primary purpose of the HFIR is the production of transuranium elements, its usefulness is increased by a number of experimental facilities which are located in, or terminate in, the reflector. These include three horizontal beam tubes, four slant access tubes called engineering facilities, and 38 vertical holes.

5.6.1 Horizontal Beam Tubes

The reactor is provided with three nominally 4-in.-ID horizontal beam tube experimental facilities which extend outward from the reactor core at the midplane (elevation 820 ft 6 in.), as shown in Fig. 5.6.1. One beam tube, HB-2, extends radially from the reactor center line, with its inner end penetrating the permanent reflector. Another tube, HB-3, extends tangentially from the core, offset approximately $10\frac{1}{2}$ in. from the reactor center line. It also penetrates the permanent reflector. The remaining tube is aligned on a tangential line approximately $15\frac{1}{8}$ in. from the reactor center line with both ends extending outward from the reactor. It is arranged to allow the installation of either two individual facilities or a single through tube. The two ends of this tube are designated HB-1 and HB-4. The beam tubes themselves are of GS 11A-T6(6061-T6) aluminum and in the high-pressure region are designed in accordance with ASME Boiler and Pressure Vessel Code, Unfired Pressure Vessels, Section VIII, 1962 edition.

Each of the tubes is sealed to, and supported by, the reactor pressure vessel by a system of clamped and bolted flanged joints. From the flanged connection at the pressure vessel, each tube continues through the reactor pool and pool wall and terminates in a recess located in a large cavity in the reactor pool wall, where it is fastened to the outside of the pool wall by a bolted flanged joint. This joint is sealed to a continuation of the pool liner by a double-bellows, flexible joint.

Because of the double-bellows joint, the outer ends of the beam tubes are free to move axially, to compensate for changes in pressure and temperature, but are fixed to prevent movement in other directions to maintain alignment. That portion of the tubes which traverses the pool and pool wall may flex freely to absorb vessel movements which may occur. The inner ends float freely in recesses in the reflector and have sufficient clearance to prevent stresses due to differential expansion. The through-tube assembly extends continuously without joints through the pressure vessel. To accommodate expansion and contraction, the flanged joints which connect the through tube to the nozzles of the pressure vessel are flexibly sealed by stainless steel bellows expansion joints. Each of the beam tubes is provided with a relief valve which vents to the reactor pool to relieve pressure inside the tubes in the event of a leak from the high-pressure region.

Each of the beam tubes passes through the pool wall and terminates in a recess located in a large beam port cavity in the shield wall. A stainless steel, stepped sleeve cast in the concrete provides access through the wall, which at this point is of 4-ft-thick high-density concrete. This 14-in.-OD sleeve is stepped to 18 in. OD at the outer end. Stainless steel shield blocks surrounding the outer end of the beam tube fill the stepped region. The remaining voids are completely filled with water, and the sleeve forms an extension of the pool liner to which the outer beam tube flange is sealed.

The recess allows the installation of a shutter or stacked shielding. Additional rollaway shielding or stacked block will be provided in and outside the cavity. The flared portion of the cavity has an acid-resistant paint applied to the concrete surfaces, and carbon steel angles are provided at all exposed corners. The remaining portion is lined with $\frac{1}{4}$-in. carbon steel plate. The cavity is $6\frac{1}{2}$ ft high and flares in width from 3 ft at the beam port face to 13 ft 9 in. at the outer face of the reactor pool wall. Rails covered with steel grating are provided in the floor at each beam port to facilitate movement of shielding.

The recess inside the cavity is provided with a $2\frac{1}{2}$-in. SBHE vent, a $1\frac{1}{2}$-in. drain, stainless steel pipe sleeves, conduit sleeves, and a 110-v duplex instrument and normal-power receptacle. These services are brought out to valve stations and to quick-disconnect couplings at the shield face beside the beam port. Other services at each beam port include 60 psig air, demineralized water, reactor pool water (supply and return), process drain, and an open hot off-gas (OHOG) vent line. Additional electrical outlets are also provided.

A stepped hole 18 in. from the inner edge of each beam port recess extends vertically from the beam port cavity through the reactor pool shield and terminates in the experiment room floor at elevation 833 ft. These holes are lined with carbon steel and are stepped from 6 to 8 in. OD. They permit a vertical beam to be taken from the beam port, or they may be used for other access to this region. When not in use they will be plugged. Shielding is provided as required when they are in service.

At the reactor end, each of the beam tubes is force-convection cooled by water flowing through 24 parallel milled grooves cut longitudinally into, and equally spaced around, the periphery of the tubes. Defined flow passages are formed by covering the tube with a close-fitting jacket. Driving force for the flow is obtained by utilizing the hydraulic pressure drop through the reactor core. As shown in Fig. 5.6.2, water flows into these passages through annular spaces between the jacket tube and the reflector. It flows outward to the flanged joint, where the tube is attached to the pressure vessel nozzle, reverses direction, flows through an annular passage outside the jacket to a collection ring inside the pressure vessel, and then to a line which discharges into the low-pressure region beneath the reactor core.

5.6.2 Engineering Facilities

Provision is made for the future installation of up to four engineering facilities to accommodate experiments which require a relatively low neutron flux. These facilities consist of 4-in.-OD tubes which enter the pressure vessel at elevation 825 ft and extend downward at an angle of approximately

41° to provide access to the flux at the outer periphery of the beryllium. The upper ends of the tubes terminate at the outer face of the pool wall in the experiment room. The proposed arrangement is shown in Fig. 3.3.3. These facilities are similar in structural details to the radial beam tubes except that cooling of the high-pressure region may be provided by natural convection. Appropriate nozzles on the pressure vessel and lined access holes through the pool wall are included in the system design.

Pumps, heat exchangers, and other such equipment required for loop experiments or similar equipment for other types of experiments can be housed as needed in the experiment room. A variety of service connections are provided near the point where each of the experiment facilities penetrates the pool wall. These include instrument air, normal and normal-emergency electric power, process water, pool water, CHOG and OHOG outlets, and both process-waste and intermediate-level-waste drains. The entire experiment room is served by the special building hot exhaust system which can also serve future experiment cells as required.

5.6.3 Irradiation Facilities

The permanent reflector is penetrated by 22 vertical holes which extend completely through the beryllium. The vertical holes are lined with 6061 aluminum. Sixteen holes have an inside diameter of 1.584 in. They are located concentric with the core and on two circles of radii 15.437 and 17.344 in. respectively. The other six holes have a 2.834 in. ID and are located on a concentric circle of radius 18.219 in. In addition, 16 similar but unlined $\frac{1}{2}$-in.-ID vertical holes are provided, eight in the removable beryllium and eight in the control plate access plugs. The general arrangement is shown in Fig. 5.1.3.

These vertical holes provide space for various types of static irradiations. In addition, it is anticipated that several of them will be utilized as locations for future hydraulic facilities which will permit the insertion and removal of samples during reactor operation. When not in use, the vertical facilities will be filled by beryllium plugs which will be provided with adequate cooling passages.

The vertical facilities in the permanent beryllium are located beneath the upper plenum cover. Access to these facilities is by means of special penetrations through the plenum cover. The vertical facilities in the control plate access plugs may be reached by removing the inner shroud which is also necessary for fuel replacement. Those in the removable reflector are accessible through removable plugs in the upper shroud flange.

A hydraulic tube penetrates the quick-opening hatch and goes into the central target region, occupying the position otherwise filled by the central target rod. The system is designed to the same criteria as the reactor vessel and utilizes the primary coolant for cooling and driving the irradiation capsules. The tube can contain up to nine $\frac{1}{2}$-in.-diam, $2\frac{9}{16}$-in.-long capsules at a time. The usual control valves and flow monitors are located in the east pool.

6. COOLING SYSTEMS

6.1 Introduction

The cooling requirements for power operation of the HFIR are satisfied by two separate cooling systems. One of these, the reactor cooling system, is designed to remove virtually all the energy from the core. In this system demineralized water as the primary coolant is pumped through the reactor tank at a flow rate of about 16,000 gpm. It then passes through the tube side of three of the four primary heat exchangers; at this point it gives up its heat to the secondary coolant, which is circulated through the shell 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 0.3 Mw of reactor heat[1] will be transferred to the reactor pool by conduction from heated surfaces and as the result of absorbed radiation. To accommodate this and up to 0.5 Mw of heat released by stored spent fuel elements, a second cooling system, the pool coolant system, permits circulation of 1000 gpm of pool water through the shell side of two heat exchangers. The same secondary coolant system is used as in the case of the reactor cooling system.

Because of the heat generated by the fission product inventory in the core, it is necessary to provide cooling to the core for some 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 motors can provide coolant flow sufficient to prevent damage due to afterheat. This dc-powered circulation system can also be used for operation at 10 Mw during a normal-power outage.

In association with each of the water systems are various components for demineralizing, deaerating, and flow measuring. Most of these components are located in the water wing of the building.

6.2 Primary Coolant System

The reactor primary coolant is demineralized light water maintained at a pH of 5.0 $\pm$0.1 to minimize formation of an insulating oxide film on the aluminum-clad fuel element. The system is designed to operate at a maximum reactor inlet pressure of 1000 psi; however, the normal operating condition is with an inlet pressure of 650 psi.

As illustrated in Figs. 6.2.1 and 6.2.2, the primary coolant enters the pressure vessel through two 16-in. inlet lines located near the top of the vessel. It passes in parallel flow paths through the fuel elements, control region, target, and reflector and leaves the pressure vessel through an 18-in.-diam line at the bottom of the vessel. The inlet temperature is usually maintained at 110°F, although the design provides for operation at 120°F. At 100-Mw operation with 16,000-gpm flow, the exit temperature is 154°F, and the core pressure drop under these conditions is about 110 psi.

---

[1]N. Hilvety, L. A. Haack, and J. R. McWherter, *HFIR Pool Criteria*, ORNL-CF-61-3-82 (March 1961).

From the pressure vessel, the exit line passes through the pipe tunnel, where it branches into four parallel headers that lead the hot water to the four heat exchanger cells (Fig. 6.2.3). The 110°F water leaving each heat exchanger enters the associated main circulation pump. Flows from the circulation pumps combine in the pipe tunnel and return to the reactor vessel through a strainer. Sensing devices for measuring temperature and pressure, as well as the main venturi for measuring flow, are located in the return line within the pipe tunnel.

A 200-gpm bypass flow is taken from the main coolant flow stream at the heat exchanger exits. This bypass flow is sent to the primary coolant cleanup system, which consists of a deaerator, filters, and a demineralizer. Primary coolant pressure is maintained by pressure-controlled letdown valves located in these bypass lines at the heat exchanger exit. The 200-gpm return flow of clean water passes through the primary coolant pressurizer pump and returns to the main system in the pipe tunnel just upstream of the monitoring devices. This return flow is adjusted manually by varying the pressurizer pump rotor speed using a magnetic coupling. The high-pressure portion of the system is provided with three relief valves: one, set at 25 psi below design pressure, discharges to the primary coolant letdown header; and two, set at 25 psi above design pressure, discharge to the primary coolant storage tank.

It is convenient to divide the primary coolant system into two subsystems: (1) the high-pressure system, which has a capacity of approximately 11,000 gal and includes the reactor pressure vessel, high-pressure piping, primary heat exchangers, pressurizer pumps, and main coolant pumps and (2) the low-pressure system, which has a capacity of approximately 7000 gal and includes the cleanup system deaerator, filters, and demineralizers. The systems are separated by the letdown valves and the pressurizer pumps.

6.2.1 High-Pressure System

(a) Reactor Vessel. – The reactor vessel, described in detail in Sec. 5, is designed to operate at internal pressures up to 1000 psi at temperatures below 200°F. It is fabricated of $2\frac{7}{8}$-in. thick carbon steel with 0.1-in.-thick weld-overlaid stainless steel on the outside and $\frac{1}{8}$-in.-thick stainless steel cladding on the inside. The vessel top head is 14 in. thick, also of stainless-steel-clad carbon steel. Drive components for control plates and fission chambers enter through a bottom head. Various slant, vertical, and horizontal openings are provided for experiment installation. The top head includes a central quick-opening hatch through which fuel elements, targets, reflector pieces, and control plates can be moved. In addition, there are numerous flanged openings in the top head to permit access to the vertical experiment positions in the beryllium.

(b) Piping and Strainer. – High-pressure stainless steel piping connects the reactor vessel with the heat exchangers and pumps. Water enters the top of the reactor vessel through two diametrically opposed 16-in. lines. The outlet from the reactor vessel is a single 18-in. line which runs from the bottom of the tank through the pool liner and biological shield to the pipe tunnel. From this point the header continues in the tunnel to feed the four individually compartmented heat exchanger–pump combinations. A 10-in. inlet line branches from the main header to the tube side of each heat exchanger. From the heat exchanger outlet, the line continues to the main circulation pump and then to a 20-in. return header in the pipe tunnel. This 20-in. return header runs parallel to the outlet header in the pipe tunnel and continues into the reactor pool. Here it passes through the inlet strainer, which contains a wire-mesh basket with a maximum opening of 0.023 in., before branching into the two 16-in. reactor inlet lines and entering the reactor vessel. The general arrangement is shown in Figs. 6.2.3 to 6.2.5.

(c) Heat Exchangers. – The primary heat exchangers are of the shell and U-tube type, mounted vertically and designed to permit tube-bundle removal. Each heat exchanger, together with its associated circulation pump and letdown valve, is located in an individually shielded compartment. Each heat exchanger is designed to transfer ~117 million Btu/hr ($34\frac{1}{3}$ Mw) from the primary coolant loop to the secondary coolant loop. Thus, only three heat exchanger–pump combinations are required for full-power operation, The nominal design flow rates are 5000 gpm at 167°F in the primary system and 6667 gpm at 85°F in the secondary system, for each heat exchanger.

The heat exchangers are ~34 ft long and contain 1190 U-tubes, each $^5/_8$ in. in diameter. The tube side is designed to operate at up to 1000 psi at 200°F, and the shell side is designed for 150 psi at the same temperature. The tube bundles can be removed by the reactor bay crane through normally plugged and shielded hatches in the top of the heat exchanger cell (reactor bay floor). The heat exchanger support channel is at the 826-ft level, with the remainder of the heat exchanger extending into the pit to the 796-ft elevation. All primary coolant system valves are located in the heat exchanger cells. Each heat exchanger–pump combination can be isolated from the system by motorized valves on the heat exchanger inlet and pump discharge lines. Personnel may enter the cell containing an isolated unit while the reactor is at power. Access to the cells is from the first floor of the water wing.

(d) Circulation Pumps. – The vertical shaft centrifugal primary coolant pumps take their suction from individual heat exchangers. Each pump is located in the same cell as its associated heat exchanger. The hatches in the reactor bay floor permit pump or motor removal by the reactor bay crane. Each main pump (PU-1A, -1B, -1C, -1D) will deliver about 5000 gpm against a 365-ft head of water when three pumps are operated in parallel. They are driven by 2300-v, ac, 600-hp squirrel-cage induction motors coupled to the pump shaft (Fig. 6.2.6).

Directly coupled to each ac motor shaft is an auxiliary 3-hp series-wound dc motor called a "pony motor." These motors are supplied with 120-v power from a failure-free system (see Sec. 10), and each one is capable of supplying 1300 gpm to the reactor. Shaft seals for the pumps are of the limited-leakage mechanical type. Curves showing the pump characteristics are given in Fig. 6.2.7.

In normal operation, both the main ac motor and the pony motor are energized and supply torque to the pump rotor. Because only three main pumps are normally required at full power, a selector switch in the control room is provided to permit selection of the spare ac motor, which is not to be included in the automatic startup sequence. The corresponding pony motor must be locked out by means of a manual circuit breaker. The start-stop pushbuttons and "running" lights for the main pump motors are located in the control room, and a stop pushbutton is located outside each of the pump cells. The main pump motors are automatically stopped by any one of the following conditions:

1. overload (relay must be hand reset in order to restart),
2. ground fault (relay must be hand reset to order to start),
3. opening of the door to the 2400-v motor starter,
4. excessive vibration,
5. low pressure in the primary loop,
6. loss of 2.4-kv normal-power.

Whenever a main circulation pump is on the line, the associated pony motor will be energized. The pony motor may be shut down only by opening a safety switch located in the battery room. An ammeter in the metering cabinet indicates current supplied to, or taken from, the battery bank. The dc pony motor also has a shunt field which permits a test at full load current with the main motor in operation. The test push button which energizes the shunt field is located in the control room, as is the ammeter indicating the test current. The test circuit is shown in Fig. 6.2.8. The test is prevented on loss of ac power to the main motor.

Alarms are received in the control room should any of the following abnormal conditions exist at the primary coolant pumps:

1. winding temperature higher than normal,
2. winding temperature approaching insulation damage point,
3. excessive vibration,

4. high motor bearing temperature,
5. charger failure or dc ground in battery charger,
6. abnormally high current to the pony motor series field,
7. abnormally low current to the pony motor series field,
8. low injection water flow to the mechanical seals.

(e) Pressurizer Pumps. – The two main pressurizer pumps, PU-4A and PU-4B, are nine-stage horizontal shaft centrifugal pumps. A variable-speed drive enables each pump to deliver up to 300 gpm at pressures up to 1000 psi. The characteristics of these pumps are shown in Fig. 6.2.9. A small 30-gpm centrifugal pump, operated by the normal-emergency power supply, maintains primary loop pressure and circulation pump seal injection water flow during a failure of the normal-power. The pressurizer pumps are located on the ground floor of the water wing. They take water from the primary water head tank (see Sec. 6.2.2) and discharge into the high-pressure system between the main circulation pumps and the inlet to the reactor vessel.

The flow rate of water bypassed into the low-pressure cleanup system is determined by the speed of the pressurizer pumps. The set point of the pressure-controlled letdown valves controls the pressure in the primary system. The pressurizer pump variable-speed unit consists of a 2300-v 300-hp squirrel-cage induction motor and an "eddy current" coupling. A selector switch in the control room permits the selection of a pressurizer pump to be started in the "automatic sequence" mode. Each pump has a spring return "stop-neutral-start" selector switch, a standby push button, and a "running" light in the control room. In normal operation one pump is running while the other is in standby. The standby pump will automatically start upon an abnormal reduction of primary coolant system pressure. The pressurizer pumps are automatically stopped by the following conditions:

1. overload (relay must be hand reset to start),
2. ground fault (relay must be hand reset to start),
3. opening of the door to the 2400-v motor starter,
4. low coolant flow to the variable-speed coupling unit,
5. loss of voltage,
6. low level in primary coolant head tank.

The eddy-current coupling variable-speed drive units are automatically started and stopped whenever the corresponding drive motor is started and stopped. Each unit is provided with a manually operated potentiometer and a speed indicator; both are located in the control room.

The emergency pressurizer pump has a locally mounted "run-off-auto" switch to permit selection of the mode of operation. In the automatic mode the pump is automatically started by a low-flow sensing element in the discharge line of the main pressurizer pumps or by an auxiliary contact in normal-emergency switch-gear unit No. 1 during a failure of normal-power. Once started, the pump will continue to run until manually stopped. "Running" lights for the pump are located at the local "run-off-auto" switch and at a "remote-control off" selector switch in the control room, from which the pumps may be shut down. When either the "remote-control off" switch or the "run-off-auto" switch is in the "off" position, an alarm is sounded in the control room.

6.2.2 Low-Pressure System

The low-pressure portion of the primary cooling system is the primary cleanup system, separated from the high-pressure cooling loop by the letdown valves and the pressurizer pumps. This system contains the following equipment: deaerator, pumps, prefilters, demineralizers, afterfilters, and primary coolant head tank with interconnecting piping (Figs. 6.2.10 and 6.2.11). The arrangement of this equipment, located in the water wing, is shown in Fig. 6.2.12.

(a) Description of Flow. – Primary coolant discharged through the letdown valves is sent through the primary coolant cleanup system before being returned to the high-pressure system by the pressurizer pumps. The nominal design flow rate through the cleanup system, controlled by varying the pressurizer pump speed, is 200 gpm. The flow from the letdown valves is joined by small flows of primary coolant from the N^{16} monitoring system, the primary circulation pump seals, the reactor tank top vent (to remove trapped gases), the cladding failure detector, and the hot-water injection system. The composite of these flows passes through a 48-ft section of 12-in.-diam pipe located in the pipe tunnel. At a flow of 200 gpm this provides ~90 sec for the decay of N^{16} before the water leaves the pipe tunnel. From the N^{16} decay line, the water enters the primary coolant deaerator. Here most of the dissolved gases are removed and sent to the closed hot off-gas (CHOG) system. The deaerator also serves as a surge tank to handle changes in the letdown flow rate.

One of two 200-gpm centrifugal pumps, PU-2A or PU-2B, pumps water from the deaerator through the remainder of the cleanup system to the primary coolant head tank. A control valve throttles the flow from these cleanup pumps to maintain a constant level in the deaerator. The cleanup flow passes through one of the two prefilters into one of the two demineralizer systems. These systems are identical; each consists of a cation bed followed by an anion bed. The common discharge line from the demineralizers passes through the afterfilter before discharging into the primary coolant head tank. An afterfilter located in the exit line from the demineralizers prevents the resin from escaping.

A level control valve supplies plant demineralized water to the primary coolant head tank during primary coolant cooldown or as required by leakage. The primary coolant head tank provides the suction head for the pressurizer pumps.

(b) Deaerator. – The primary coolant deaerator, shown in Fig. 6.2.13, is designed to reduce the concentration of dissolved gases in the primary cooling water. During normal operation these include Ar, O_2, H_2, CO_2, and traces of fission gases. In the event of a fuel element meltdown or leak, the gaseous fission products are removed from the water and discharged into the CHOG system. The deaerator and its associated steam jets and condensers are located on the first floor of the water wing. This equipment is enclosed in a cell shielded by 3 ft of high-density concrete or its equivalent. Inlet and exit primary water lines and off-gas lines are also shielded.

Primary water enters the deaerator tank from the N^{16} decay line located in the pipe tunnel. As the water enters the deaerator, it falls through a bed of Raschig rings and is collected in the bottom of the tank. The primary cleanup pumps, PU-2A and PU-2B, take their suction from the bottom of the deaerator vessel. A pneumatic level sensor transmits a deaerator vessel liquid-level signal to the control room and to the flow control valve located on the discharge side of the cleanup pumps. The pH is monitored by a cell which receives a sample from the common discharge of these pumps and returns it to the pump suction. Water conductivity is also monitored at this point. Both conductivity and pH are recorded in the control room.

A vacuum is maintained in the deaerator tank by a steam-jet ejector system which consists of a precondenser, high-vacuum ejector, low-vacuum ejector, and an aftercondenser. The evolved gases are discharged to the CHOG system. The condensers are cooled by secondary coolant. Condensate from both pre- and aftercondensers is returned directly to the deaerator tank. A deaerator overflow line is provided which discharges into the intermediate-level-waste system (see Sec. 11.3.2).

(c) Filters and Demineralizers. – The primary coolant filters and demineralizers are designed to remove particulate and dissolved material from the primary coolant. During normal operation the prefilters prevent particulate material from reaching the demineralizers. The demineralizer resins remove dissolved contaminants from the water. The afterfilter removes demineralizer-resin fines from the water before it is returned to the high-pressure system. In the event of a fuel meltdown, particulate and dissolved fission products in the water will be removed by the filters and demineralizers. The prefilters, demineralizer units, and afterfilter, shown in Fig. 6.2.13, are located in a pair of shielded cells on the ground floor of the water wing.

Each of the two prefilters has a nominal flow capacity of 200 gpm and consists of 28 porous tubes ($2\frac{3}{4}$ in. in diameter by $19\frac{1}{2}$ in. long) made of sintered stainless steel. Water flows from the outside to the inside of the tubes through 20-μ pore openings. These filter elements can withstand a pressure drop of 150 psi; however, the pressure drop across a clean filter element is 2.5 psi. Additional shielding is provided by 3 in. of lead around the filter shell. Both prefilters are vented to the open hot off-gas (OHOG) system through ball-float traps. Each unit can be remotely flushed and cleaned from outside the shielded cells.

The demineralizer system consists of two units in parallel, each capable of a flow rate of 200 gpm. Each unit consists of a cation bed and an anion bed in series. Each cation bed contains 35 ft^3 of cation resin in a vessel 4 ft in diameter by 6 ft high. Each anion bed contains 60 ft^3 of anion resin in a similar vessel 5 ft in diameter by 6 ft high. Under normal operating conditions, one unit is in service. Some of the cation effluent may be bypassed around the anion bed to aid in maintaining a system pH of 5.0 $\pm$0.1. Additional local shielding is provided by 3 in. of lead surrounding the cation tanks. The anion tanks are not directly shielded.

The afterfilter is designed for a 200-gpm flow rate. The filter element is 100-mesh stainless steel screen with a mean pore opening of 120 μ. The filter element can withstand a maximum pressure differential of 150 psi, the total discharge head (TDH) of the cleanup pumps. Pressure drop across a clean filter is 1.5 psi at a 200-gpm flow. The afterfilter can be operated, bypassed, and cleaned from outside the shielded cell.

From the afterfilter the clean water is sent to the primary coolant head tank, located on the first floor of the water wing. It is a 2500-gal horizontally mounted stainless steel tank 6 ft 6 in. in diameter by 11 ft 8 in. long.

The pH is controlled by a system which adds nitric acid to the primary coolant head tank as required to maintain the high pressure system at pH 5.0.

6.2.3 Makeup, Fill, and Drain Systems

During routine operation at pressure, makeup water to replace that lost by leakage from the system is furnished by the plant demineralized water pumps, PU-18A and PU-18B. This is automatically supplied to the primary coolant head tank through a water level control valve.

Several lines are available to permit filling, draining, and flushing of the system at low pressure, as shown in Fig. 6.2.14. A shielded 20,000-gal primary coolant storage tank is located underground at the northeast corner of the building. The entire contents of the primary coolant system can be sent directly to this tank or can be routed through the primary coolant demineralizers by means of a 2-in. line which bypasses the pressure letdown valves and deaerator. A 3-in. line connects the discharge of the primary coolant afterfilter to the discharge line of the pressurizer pumps, permitting the primary coolant cleanup pumps to send water to the primary coolant system. Thus, it is not necessary to use the large pressurizer pumps to circulate primary coolant through the filters and demineralizers while the reactor is depressurized.

The roof top of the underground water storage tank is at grade level. Its walls, floor, and roof are of 14-in.-thick concrete. The inside is lined with plastic sheets bonded together to form a watertight shell. A 4-in. line exits from the bottom of the storage tank, enters the east side of the water wing, and leads to the primary coolant cleanup pumps. The transfer line can be valved into the suction or discharge of either or both these pumps to transfer water to and from the underground storage tank.

6.3 Secondary Coolant System

6.3.1 Introduction

Heat from the HFIR complex is dissipated to the atmosphere by a conventional induced-draft cooling tower. The secondary coolant pumps circulate water through the cooling tower and the various heat exchangers, thus removing heat from the different systems. Of the nominal 26,000 gpm circulated in the system, ~20,000 gpm passes through the primary coolant heat exchangers, ~4200 gpm through the auxiliaries in the water wing, and ~1800 gpm to the adjacent Transuranium Processing Facility (TRU). The secondary coolant system is shown in Figs. 6.3.1 to 6.3.3.

A fire hose connection is available to supply emergency secondary coolant.

6.3.2 Cooling Tower

The cooling tower is a four-cell induced-draft tower located southeast of the reactor building. It is designed to transfer 375 million Btu/hr by cooling 26,000 gpm of 115°F water to 85°F at an ambient wet bulb temperature of 77°F. Of this total heat transferred, 350 million Btu/hr is supplied by the primary coolant, 10 million by the pool coolant, 8 million by the air-conditioning system, 3 million by other loads, and 4 million is reserved for future operations. The cooling tower is composed of four individual cells mounted as one unit over a concrete basin approximately 120 ft long by 54 ft wide. The basin is divided into four separate compartments, each with a capacity of 100,000 gal. This 400,000-gal storage capacity is sufficient for ~8 hr of full-power operation without makeup water. Each compartment can be drained and cleaned individually while the rest of the tower is operating. Potable water is supplied to the tower basin through a float valve to make up for evaporation, drift loss, and blowdown.

6.3.3 Cooling Tower Fans

Each cell is equipped with two 50-hp two-speed fans which can be reversed for deicing the tower. Operation of the cooling tower fans is controlled by a "run-off-auto" mode switch on the control room process panel board. When in the run mode, each fan is controlled by a three-unit push-button station located in the control room. This station provides alternatively for (1) fast speed–forward direction, (2) slow speed–forward direction, and (3) slow speed–reverse direction.

A time-delay relay provides for automatic deceleration when transferring to slow speed forward. Mechanical interlocks permit only one starter coil to be energized at a time. The different speeds are designated by lights located on process control panel E in the control room.

In the automatic mode the fan speed is controlled by a temperature controller through sixteen switches. Each of these switches operates time-delay relays for fast forward speed and for slow forward speed. The time-delay relays are adjustable and perform the following functions:

1. Provide different time settings for the relays, allowing the fan load to be added to the motor control center in a stepped sequence. This limits the voltage drop to permissible levels when voltage is restored under maximum cooling requirements following a power outage.
2. Limit the number of starts, stops, and speed changes by adjustment of the time delay, thus preventing motor overheating due to cycling.
3. Permit (by the relays) the use of low-voltage switches to operate the high-voltage fan starters.

In the automatic mode the fans will automatically restart after a power failure. A typical response to an automatic startup is shown in Fig. 6.3.4. In the run mode the fans must be restarted manually following a power failure. All the fans are automatically shut down by a fire alarm from the cooling tower. A vibration switch near each fan shuts down only the affected fan. These devices must be manually reset before the fans can be restarted.

6.3.4 Water Treatment

The blowdown and chemical treatment of the secondary coolant system are designed to inhibit corrosion, scale formation, and microbiological growth. The water in the secondary coolant loop is chemically treated by (1) automatically regulated addition of H_2SO_4 to control pH; (2) automatic addition of phosphate or chromate chemicals to inhibit corrosion; and (3) manual addition of non-oxidizing biocides to control algae.

The acid-injection system consists of a 5000-gal H_2SO_4 storage tank, two proportionating chemical feed pumps, an acid-water mixer, and an instrumentation system which measures pH and controls the pumps. Phosphate-chromate treating equipment consists of a dissolving tank and a pair of manually adjustable proportionating feed pumps.

6.3.5 Pumps

The three main secondary coolant pumps (PU-6A, 6B, and 6C) are vertically mounted centrifugals with a combined capacity of 26,000 gpm. The characteristics of these pumps are shown in Fig. 6.3.5. A smaller fourth pump (PU-14) has a capacity of 6000 gpm and provides shutdown and emergency water circulation. These pumps are mounted in a separate pump basin along the north side of the cooling tower. The pump basin is fed by a flume which collects water from the individual tower basins. A chemical treatment distribution header above the pump basin mixes the chemicals with the basin water.

During periods when the reactor is shut down, the cooling demand is greatly reduced. The 6000-gpm pump is then put into service, and the main pumps are stopped. This pump maintains cooling for the auxiliary systems. It also provides an emergency flow of 3000 gpm during a normal-power outage when a second winding receives power from the auxiliary diesel-motor-generator set.

A selector switch in the control room permits choice of a main secondary pump to be excluded from the automatic startup sequence. A spring return "start-neutral-stop" switch for each pump is located in the control room. The pumps are automatically stopped by the following conditions:

1. overload (relay must be hand set to restart),
2. ground fault (relay must be hand set to restart),
3. opening of 2400-v motor starter door,
4. normal-power failure.

The auxiliary 6000-gpm circulation pump, PU-14, is driven by a two-winding motor. The high-speed winding provides sufficient secondary coolant flow (6000 gpm) to handle the normal requirements at the site when the reactor is not operating at power. The slow-speed winding, supplied from the normal-emergency power system, provides sufficient cooling (3000 gpm) to permit reactor operation at 10 Mw during a normal-power outage because other heat sources are inoperative. A spring return "stop-neutral-start" switch for the high-speed winding and a "running" light are located in the control room. Electrical and mechanical interlocks between the high- and low-speed starters permit only one to be energized at a time. A "run-off-auto" mode switch and a "running" light are located in the control room. This permits selection of the mode of operation for the slow-speed winding and indicates the mode chosen. An alarm is sounded in the control room when the switch is placed in the "off" position.

In the automatic mode the slow-speed winding is energized upon loss of normal-power by a contact in normal-emergency transfer switch No. 1. The contact is closed when the transfer switch is in the emergency position (i.e., the diesel generator is supplying power to the No. 1 normal-emergency circuit).

After normal-power has been restored, the pump will continue to run until manually shut off. The slow-speed starter is interlocked with a solenoid valve so that return flow is routed through a bypass directly to the tower basin when this winding is energized.

The secondary coolant pumps discharge water into a 42-in. pipe which runs underground to the east side of the reactor building, where it divides into a 36- and an 18-in. line (see Fig. 6.3.3).

The 36-in. line carries approximately 20,000 gpm of cooling water to the primary coolant heat exchangers. The 18-in. line supplies the pool heat exchangers, air-conditioning units, other auxiliary equipment, and the TRU facility. Return water lines to the cooling tower run parallel to the supply headers. Remotely operated motor-driven control valves on the inlet and exit of each primary heat exchanger throttle the flow to the individual heat exchangers and permit them to be isolated for cleaning and repair. An automatic flow control valve regulates the flow of secondary coolant to the heat exchangers, as required to maintain a constant primary coolant temperature at the reactor inlet.

6.4 Pool Water Systems

The reactor pressure vessel is located in a cylindrical pool 18 ft in diameter and 36 ft deep which contains approximately 80,000 gal of water. The general arrangement is shown in Figs. 3.1.1 and 6.2.4. Connected to this pool is a rectangular storage pool, $41\frac{1}{2}$ ft long, 20 ft deep, and 18 ft wide, which contains approximately 114,000 gal of water. A smaller cylindrical pool 8 ft in diameter and 25 ft deep is located at the east end of the rectangular pool. This is provided to accommodate a future critical assembly. The critical assembly pool contains approximately 10,000 gal of water.

The reactor pool is separated from the rectangular pools by a removable gate. The rectangular pool, called the clean pool, is divided into two 20-ft sections by a second removable gate. All the pools are lined with stainless steel to prevent leakage, to facilitate decontamination, and to help maintain the water purity.

The pool water requires circulation and heat exchange to dispose of up to 0.8 Mw of heat absorbed from the reactor gamma radiation, hot primary coolant system components, and stored spent fuel elements. Moreover, to reduce corrosion and to maintain a low contamination level, it is necessary to circulate a fraction of the pool water through a cleanup system. Flow diagrams are shown in Figs. 6.4.1 and 6.4.2.

6.4.1 Pool Coolant System

All the pools are provided with overflow scuppers at the 848-ft level. Pool water overflowing into these scuppers flows by gravity into the pool surge tank, as shown in Fig. 6.4.3. The pool surge tank maintains a positive suction head on the two pool coolant pumps, PU-9A and PU-9B, each rated at 1000 gpm. Coolant flow leaving the pumps goes through the pool filter and then enters the shell side of the pool heat exchangers. Either exchanger can handle up to 1000 gpm. Secondary coolant entering the tube side of the heat exchangers is automatically regulated by a valve to maintain the preset temperature of the pool water leaving the exchangers.

From the heat exchangers the cooled water may be diverted to the various pools as required by the heat load distribution. Various temperature sensors, located in the overflow lines from the pool scuppers, indicate the temperature distribution in the pools.

The system is balanced to achieve the desired flow distribution between the pools by manually adjusting the valves in the return lines to the pools. A level control valve supplies plant demineralized water to the surge tank as makeup for evaporation and leakage.

The pool coolant pumps also serve as pool water transfer pumps. They can transfer water to and from the underground storage tank.

(a) Pool Overflow. – The reactor pool overflow of approximately 500 gpm flows by gravity through two lines to the 8-in.-diam common pool overflow collection line. Approximately 400 gpm from the clean pool and 100 gpm from the critical pool also flow into this line. The collection line dips below the level of the pool surge tank before entering it, thus providing a water seal for the scupper drains.

The temperature of the water overflowing from each pool is monitored by thermocouples and is displayed in the control room. A float switch in each pool monitors the water level, actuating an alarm in the control room if an abnormally high or low level is detected. The pool coolant pumps are automatically shut off if a high water level is detected in the reactor or critical pool. A radiation monitor is located in the common collection line; the activity level of the water is recorded in the control room. An alarm is also received in the control room if a high-radiation level is detected by this instrument.

(b) Pool Surge Tank. – The pool surge tank, which collects the overflow from the pools, has a capacity of 1250 gal. It is located on the first floor of the water wing at the west end of the pool demineralizer cells. A water level sensor transmits the water level indication to the control room. If a low water level is detected in the surge tank, an alarm is received in the control room, and the pool coolant pumps are automatically stopped. The level sensor also controls an automatic make-up valve which permits demineralized water to flow into the tank in order to maintain a minimum level. Any overflow runs to the process waste drain. The overflow line is equipped with a 6-ft leg to provide a water seal. Other lines connected to the surge tank are (1) a vent to the OHOG system, (2) an overflow line from the pool deaerator, (3) a normally unused crosstie to the pool coolant clean-up deaerator, and (4) a line to the suction of the pool coolant pumps.

(c) Pool Coolant Pumps. – The pool coolant pumps, PU-9A and PU-9B, each have a rated capacity of 1000 gpm. They are located on the ground floor of the water wing outside the pool demineralizer pump cell. An "on-off-reset" selector switch and a "running" light for each pump are located on the process panel in the control room. A "stop-reset" button is located at each pump. The pumps are stopped automatically by a high water level in the reactor or critical pools and by a low level in the surge tank. When a pump is shut down by a level switch for more than 45 sec, it must be reset before it can be restarted. The pumps are connected to the normal-power system and will automatically restart when power is applied following a normal-power failure. Pressure gages are mounted on the inlet and exit of each pump. The pool coolant pumps normally take their suction from the pool surge tank and discharge into the heat exchangers through or around the pool water filter. They can, however, also be used for various filling and draining operations.

(d) Pool Coolant Filter. – The pool coolant filter has a nominal flow capacity of 1000 gpm. It consists of 70 porous tubes $2\frac{3}{4}$ in. in diameter by $19\frac{1}{2}$ in. long made of sintered stainless steel. Water flows from outside to inside through pore openings having a mean diameter of 20 μ. The filter is located in a shielded cell on the first floor of the water wing. A bypass around the filter may carry an excess coolant flow. The filter can be cleaned with acid, caustic, or steam by countercurrent flow. A vent at the top of the filter is connected to the OHOG system through a ball-float trap.

(e) Heat Exchangers. – Each of the pool heat exchangers (EX-2A and EX-2B) is designed to remove 5 million Btu/hr (1.5 Mw) from the system. They each have 540 fixed tubes $\frac{5}{8}$ in. in diameter. In normal operation each exchanger operates with a flow of 500 gpm of pool water but can accommodate 1000 gpm. Pool water makes two passes on the shell side of each exchanger, and the secondary cooling water makes two passes in the tubes. Normally the secondary water flow to each exchanger is approximately 1600 gpm. The temperature of the water in the common pool water exit line from the heat exchangers is monitored by a thermocouple and is displayed in the control room. A temperature-control sensing element is also located in the common exit line. This element controls the flow of secondary coolant to the exchangers in order to maintain the exit temperature constant. Each exchanger is equipped with a manually operated shell vent to the atmosphere and a shell drain to process waste. Pool water may be routed around the heat exchangers by a bypass line, which rejoins the exit line ahead of the temperature sensing elements.

The headers to the reactor and clean pools branch off the common exit from the heat exchangers which continues to the critical pool. Each header is equipped with a flow-measuring orifice which indicates, in the control room, the return flow to each pool. These headers returning to each pool contain remotely operated valves which throttle the flow from the control room.

6.4.2 Pool Cleanup System

In order to maintain high-purity water, it is necessary to pass approximately 200 gpm of pool water through the pool cleanup system, which consists of a deaerator, a prefilter, a demineralizer, and an afterfilter in series. The task of cleaning the pool water is made more complex by the possibility that one or more defective (leaking) fuel elements will be stored in the reactor pool. The system, shown schematically in Figs. 6.4.4 and 6.4.5, is designed to handle this situation but will function equally well if no defective element is present. The equipment arrangement is shown in Fig. 6.4.6.

Pool water flows through the defective fuel element storage tanks, described in detail in Sec. 11, into the pool deaerator. From here it is forced through the pool cleanup system and returns to the reactor and storage pools in common with the pool coolant flow. As indicated in Fig. 6.4.7, the normal flow path of the pool water is into the bottom of the defective element tanks, up through the element, and out the bottom through a hollow, cadmium-lined post to the pool cleanup system. Should the cleanup system flow be interrupted, cooling for the fuel element is provided by natural circulation through a small heat exchanger immersed in the pool.

Approximately 200 gpm of pool water flows from the defective element storage tanks into the deaerator. The flow is adjusted by a remotely operated valve according to a flow indicator in the control room. The deaerator removes the dissolved gases and provides surge capacity for the system. From the deaerator, water flows to the pool cleanup pumps, PU-7A and PU-7B. The discharge from these pumps is automatically regulated by a valve to maintain a constant level in the deaerator. Water leaving these pumps passes through a prefilter, an anion bed, a cation bed, and an afterfilter before returning to the reactor and clean pools. There is only a single set of pool demineralizers. The pool volume is sufficiently large so that water conditions will not change to any great extent during the time required for regeneration. During regeneration, the demineralizers can be bypassed without affecting the operation of the deaerator and filters. A radiation-detection device located in the line to the deaerator indicates (in the control room) the amount of contamination in the water leaving the defective element storage tanks.

(a) Deaerator. – The pool deaerator, shown in Fig. 6.4.8, is designed to remove essentially all the dissolved gases from the pool water at a flow of 200 gpm. Together with its associated condensers and steam ejectors, it is located in a 1-ft-thick normal-concrete-shielded cell on the first floor of the water wing. Water enters the top of the deaerator tank and passes down through a bed of Raschig rings. The high vacuum maintained by the steam-jet ejectors removes dissolved gases from the water. Gases from the deaerator pass through a precondenser, a high-vacuum ejector, a low-vacuum ejector, and an aftercondenser before entering the CHOG system. Condensate from the condensers is returned to the deaerators. A level sensor detects the water level in the deaerator and transmits a signal to a flow control valve on the discharge of the pool demineralizer pumps to hold the level constant. The demineralizer pumps are automatically shut down if a low level set-point is reached and are automatically restarted when the low level switch is cleared. They are also shut down by a high water level that lasts longer than 45 sec in the clean pools and must, in this case, be manually restarted. The deaerator level signal is also displayed in the control room. A separate level safety switch on the deaerator transmits an alarm to the control room if a high level is detected and closes a block valve in the steam supply line to the ejectors to prevent the discharge of water into the CHOG system. Each condenser is equipped with vacuum and pressure gages to aid in adjusting the vacuum. A pressure sensor transmits the deaerator vacuum to an indicator in the control room and sounds an alarm if high pressure is detected.

(b) Demineralizer Pumps. – Water from the deaerator enters the suction of the demineralizer pumps PU-7A and PU-7B and is discharged to the pool demineralizer prefilter. These pumps are located in a shielded cell on the ground floor of the water wing. Each pump has a capacity of 200 gpm. The pumps discharge into a common line containing the deaerator level control valve. Compound pressure gages are located on the suction side of each pump, and normal gages are located on the discharge side. An "on-off-reset" selector switch and a "running" light for each pump motor are located on the process panel in the control room. A "stop-reset" button is located at each pump. High water level in the clean pool lasting longer than 45 sec will shut down the pumps, and they must be restarted manually. A low level in the pool deaerator will also stop the pumps; but, in this case, a restoration of the level starts the pumps automatically.

Either, or both, pumps may be operated at one time. The motors are connected to the normal-power system and will restart automatically when power is restored following an outage.

(c) Filters and Demineralizers. – The pool demineralizer prefilter has a rated flow capacity of 200 gpm with a 2.5-psi pressure drop. It is similar to the other prefilters and contains 28 porous sintered stainless steel filtering tubes with a mean pore size of 20 μ. Water flows from the outside of the tubes to the center. The filter is located on the first floor of the water wing in the same shielded cell as the pool demineralizers. It is covered with a 2-in.-thick lead shield and is vented through a ball-float trap to the OHOG system. The filter, which is also equipped with a bypass, can be cleaned by backwashing with acid, caustic, or steam. Pressure gages are located on the inlet and outlet.

After passing through the prefilter, water enters the pool demineralizers, shown in Fig. 6.4.8, which consist of separate cation and anion beds. They are located on the first floor of the water wing in a shielded cell. The cation column contains 50 ft^3 of cation resin and is surrounded by 2 in. of supplementary lead shielding. The anion column contains 50 ft^3 of anion resin but has no supplementary shielding. Water enters the top of the cation column, passes through it, and into the top of the anion column. From the bottom of the anion bed the water flows to the afterfilter. Both demineralizer columns are vented to the OHOG system through ball-float traps. Appropriate acid, caustic, and backwash lines are provided. A demineralizer recycle pump, PU-12, is available for recycling water during regeneration. Sample taps are located on the inlet and outlet of each column. The sample lines run outside the cell to a sampling sink.

From the demineralizers the water flows to the pool demineralizer afterfilter, located in the same shielded cell as the pool demineralizers. This filter has a rated flow capacity of 200 gpm with a 2.5-psi pressure drop when clean. The filtering medium is 100-mesh stainless steel screen. Pressure gages are located on the inlet and exit lines. The filter is vented to the OHOG system through a ball-float trap. It can be backwashed to the intermediate-level-waste system using water from the pool demineralizers. Water leaving the afterfilter can be returned to any pool through lines common with those returning water from the pool cooling system.

6.4.3 Pool Fill and Drain Systems

Each section of the clean pool has one 3-in. line entering below the surface of the water; the critical pool has a single 4-in. line entering below the surface. These lines terminate at the 835-ft level (i.e., at a water depth of 7 ft). The reactor pool contains two lines which enter below the surface. One is the normal 6-in. return from the pool coolant system which terminates at the 828 ½-ft level at a water depth of 16 ½ ft (i.e., 8 ft above the core center line). These lines also serve as fill and drain lines for the pools. The second line is the 4-in. feed line to the pool deaerator from the defective element storage tanks. This is located at the 817 ½-ft level, 3 ft below the core center line.

A 55,000-gal underground storage tank has been provided to receive pool water when draining is necessary. This tank is concrete lined with plastic and is located outside the east wall of the water wing. The top of the 14-in.-thick concrete shield is flush with the finished grade at this point. A second section of this tank provides the 20,000-gal primary coolant storage mentioned in Sec. 6.2.3. Both sections of the tank are vented to the atmosphere through the SBHE filters, and the overflow is connected to the process waste system (see Sec. 10.3).

The fill and drain lines from the individual pools run to a common fill and drain header located in a pipe tunnel on the ground floor. By an appropriate system of valving, this header can be routed into the suction or discharge of either of the pool coolant pumps (PU-9A and PU-9B). Similarly, the line from the underground storage tank can be connected to either side of these pumps to fulfill any fill or drain requirement.

6.5 Emergency Cooling Requirements

It is possible to distinguish three types of emergency cooling requirements which arise from the following causes: (1) failure of a component (e.g., a pump) in the cooling systems; (2) loss of both normal 13.8-kv electrical feeders; and (3) reactor shutdown due to some cause such as multiple component failure, power failure, human error, or a combination of these events.

The occurrence of a component failure, as in case 1, is not too unlikely; however, this event has been anticipated. All vital components have standby counterparts which are available to assume the load. Such a situation would result in, at most, the necessity for a small reduction in power.

In case 2, where the normal 13.8-kv electrical supply is interrupted, presumably for only a short time, the system can still operate at 10 Mw. However, the auxiliary power systems must operate normally, and no vital mechanical components may fail concurrently.

Because of the short life and high cost of the HFIR fuel and the high rate of growth of xenon and samarium following shutdown, it is desirable to keep the reactor at a power as high as possible, even during abnormal conditions. For this reason, certain of the emergency systems have been designed to permit short-term operation at 10 Mw even though the normal-power has been interrupted. The design features incorporated to permit 10-Mw operation during this condition are as follows:

1. Two diesel-motor-generator sets share the burden of supplying emergency power to the system. Certain crucial items are included in duplicate with one unit on each diesel; others are connected to the power source through a failure-free battery system which supplies power for sufficient time even though the diesel systems do not operate.

2. In order to prevent a scram during the switching transient caused by a 13.8-kv feeder transfer or a normal-power failure, failure-free battery systems supply power continuously to vital instrumentation. A flux-to-flow ratio computer (see Sec. 8) initiates an automatic reduction in the reactor power to match the primary coolant flow and prevent flux-to-flow ratio or high-temperature scrams.

3. An emergency pressurizer pump and emergency windings for the auxiliary secondary coolant pump, both supplied by a diesel, have been provided.

4. As described in Sec. 6.2.1(*d*), each of the primary coolant pumps is equipped with a 3-hp dc pony motor supplied from a battery system. These motors are energized at all times during operation and take over the load upon failure of the main motors.

The flows developed by the pony motors are as follows:

Number of pumps operating	1	2	3
Flow (gpm) (approx)	1300	2100	2500

The sequence of automatic operations in the cooling system following a 13.8-kv power outage is as follows:

1. The primary pumps coast down and, due to the pony motors, flow stabilizes at ~2500 gpm after ~10 sec. Simultaneously, the flux-to-flow ratio computer reduces the power in an orderly fashion to ~10 Mw, the letdown block valves close, the pressurizer pumps coast down, and the secondary coolant pumps coast down.

2. Approximately 10 sec after the power failure, diesel power is available to run the emergency pressurizer pump and the emergency winding of the auxiliary secondary coolant pump, to open the cooling tower bypass valve, and to maintain charge on the various battery systems.

Following the loss of the 2400-v bus, the same sequence is followed except that the diesels need not be started. In this case, the 13.8-kv system would supply power to the normal-emergency system and the normal winding of the auxiliary pump.

The third type of emergency cooling requirement, case 3, is that in which either because of multiple component failure, power failure, or some combination of events, it is necessary to shut the reactor down. In this case, only afterheat need be considered.

The heat-generation rate as a function of time after shutdown following a 25-day operating cycle at 100 Mw is given in Fig. 6.5.1. The heat generation drops from 100 to 7 Mw virtually instantaneously at shutdown. One pony motor will supply adequate cooling following shutdown although there are normally three pony motors energized, and the coastdown of the main pump motors provides additional flow during the first 8 to 10 sec. Forced convection, supplied by one or more of the pony motors, is required for approximately 4 hr after shutdown in order to provide reasonable assurance against fuel melting. Each of the four pony motor battery banks can supply power for a minimum of 2 hr, even in the unlikely event that both diesel generators fail to start. Operation of either diesel generator can provide power to the battery chargers, thus allowing flow to be maintained indefinitely. At the end of 4 hr the heat-generation rate has dropped to <0.7 Mw and natural convection cooling is adequate.

6.5.1 Startup of Vital Pumps

As a result of a normal-power outage of the preferred 13.8-kv feeder, the large 2.4-kv pump motors will stop. When service is restored by transfer to the lower-capacity alternate feeder, overzealous operator action in restarting these motors could cause overcurrent surges which would trip protective devices in the alternate feeder, causing a further delay.

To prevent this situation and to minimize operator time away from the control console, an "auto process start" push button is located on the console. This button starts these pump motors in an orderly sequence to prevent unnecessary overcurrent surges.

No delay – One primary coolant pressurizer pump PU-4 motor

12-sec delay – Two secondary coolant pump PU-6 motors

72-sec delay – Two primary coolant pump PU-1 main motors

84-sec delay – One primary coolant pump PU-1 main motor

The standby pressurizer pump and primary coolant pump, having been designated at the beginning of the cycle, would remain idle. The dc pony motors are unaffected by this procedure. The emergency winding of secondary coolant pump PU-14 remains energized, and the cooling tower bypass valve remains open about 15 sec after restoration of normal power – long enough for restoration of normal service.

Should the reactor control system have reverted to the "start" condition (see Sec. 8), all necessary water service equipment to obtain the "run" condition is restarted by the "auto process start" button plus the normal button for PU-2 primary coolant cleanup pump.

7. CORE PHYSICS, NUCLEAR DESIGN, AND HEAT TRANSFER

7.1 Introduction

The primary purpose of the HFIR is to produce yearly "research quantities" of various transuranium isotopes, with Cf^{252} being the heaviest isotope that can be produced in significant abundance. A secondary purpose of the reactor is to provide as many additional irradiation and experimental facilities as possible without adversely affecting the production of the transuranium isotopes.

Preliminary investigations of transplutonium production from Pu^{242} feed material indicated that a thermal-neutron flux of 2 to 3 $\times$ 10^{15} neutron cm^{-2} sec^{-1} would be required in the target to achieve the desired production rate. Since such high flux levels implied high power levels, a survey study was conducted to determine what type of reactor would provide the highest thermal-neutron flux per unit of power and which in other respects would be economically and physically suitable for transuranium production. The reactor selected was the flux-trap type with light water in the flux trap (island), this being the region in which the plutonium target is to be irradiated. The power level required for the desired flux level was 100 Mw.

The high thermal-neutron flux in the island of a flux-trap reactor results from the leakage of nonthermal neutrons from the fuel region to the island, where many of the neutrons are slowed to thermal energies. The actual magnitude of the flux on a unit reactor power basis depends on many parameters, the more important of which are the island diameter and moderator, the average power density, and the length-to-diameter ratio of the core. Many nuclear studies were made, using various reactor codes and information from appropriate critical experiments, to investigate the various parameters. Results from these studies indicate that (1) light water was the best island moderator; (2) the island had an optimum diameter; (3) the fuel region had an optimum length-to-diameter ratio; and (4) the core volume should be as small as possible for the specified 100-Mw power level.

To help achieve a small core volume, the following features were incorporated in the core design: cylindrical geometry, nonuniform radial distribution of the fuel, flux suppressors at the core ends, and a symmetrical reflector control system which is supplemented by a burnable poison in the fuel. In addition, very extensive analytical and experimental studies were made in connection with the mechanical, hydraulic, and heat-removal characteristics of the core in an effort to achieve the maximum possible heat transfer surface area–to–core volume ratio. These studies, in conjunction with fabrication considerations, resulted in the selection of 0.050-in.-thick, involute geometry coolant channels and fuel plates. The plates are assembled in two concentric fuel annuli. The heat-removal analysis, which very conservatively considered detrimental mechanical, hydraulic, thermal, and nuclear deviations from the nominal to exist at the same time and place in a consistent manner, indicated that a core volume of only 51 liters was required to remove the 100 Mw of heat. The corresponding peak thermal-neutron flux in the island was found to be about 5 $\times$ 10^{15} neutrons cm^{-2} sec^{-1}, with a corresponding steady-state incipient boiling power level of about 140 Mw. With the proposed fuel loading of 9.4 kg of U^{235}, a typical fuel cycle was calculated to be about 14 days.

The following sections discuss the nuclear design and heat transfer primarily from the steady-state point of view. Additional information concerning the kinetic behavior of the system is contained in Sec. 8, particularly Sec. 8.8 and in ORNL-3573. It should be noted that this section of the report pertains mainly to design consideration rather than an analysis of actual reactor behavior. Generally speaking, the reactor does function as was intended; however, some of the numbers and curves included herein are not completely consistent with actual operating characteristics. For instance, the calculated fuel cycle for an arbitrary set of average conditions was 14 days; the actual fuel cycle being achieved is 23 days.

7.2 Selection of Reactor Type and Materials

Preliminary analyses[1-3] of the transuranium isotope production scheme indicated that to achieve a satisfactory Cf^{252} production rate – generally defined as tens of milligrams per year – a perturbed thermal-neutron flux of about 2 to 3×10^{15} neutrons cm^{-2} sec^{-1} would be required. The assumption used in the analysis was that only 100 g of Pu^{242} feed material would be available for the first year or so of reactor operation.

Investigations[4] of various types of reactors and consideration of the plutonium target design indicated that a flux-trap-type reactor, operating at about 100 Mw, could produce the necessary thermal-neutron flux in the island region (flux trap) of the reactor. Thus, a flux-trap geometry was selected, and the design power level was specified as 100 Mw. A core of this general type is shown in Fig. 7.2.1; typical radial neutron flux distributions are shown in Fig. 7.2.2; and typical axial thermal-neutron flux distributions are shown in Fig. 7.2.3.

The selection of aluminum-clad fuel plates, light-water as coolant and moderator, and a beryllium reflector was based primarily on the proposition that this would result in minimum research, development, operating, and capital costs. Furthermore, there was no indication that these materials would not be satisfactory from a nuclear and heat-removal point of view or that the use of more exotic materials would significantly improve performance in this regard.

The eventual selection of light water for the island was based on an analysis[4] that considered light water, heavy water, beryllium, and various combinations of light water and beryllium for use in the island. With and without the plutonium target in the island, the calculation showed that light water resulted in the highest thermal-neutron flux in the island per unit of reactor power.

7.3 Nuclear Design

7.3.1 General Considerations

As shown in Fig. 7.3.1, the production of Cf^{252} from Pu^{242} requires ten successive neutron captures, and thus the production rate, at least during the first year of irradiation, is quite sensitive to the thermal-neutron flux. Figure 7.3.2 indicates that for thermal-neutron fluxes less than about 4×10^{15} neutrons cm^{-2} sec^{-1} the total Cf^{252} production during the first year of irradiation is proportional to about the third power of the flux. This dependency provided a strong incentive to achieve the highest practical thermal-neutron flux in the plutonium target at the specified 100-Mw maximum power level and justified the development of several design features that significantly increased the maximum achievable neutron flux. A brief summary of typical HFIR nuclear characteristics is given in Table 7.3.1.

The high thermal-neutron flux in the island of a flux-trap reactor results from the leakage of nonthermal neutrons from the fuel region to the island, where many of the neutrons are thermalized. The diffusion length of the thermalized neutrons is less than that of the neutrons leaking from the fuel; so in a sense the neutrons are trapped in the island moderator. This description of the flux-trap principle is attributed to Wigner.[5]

---

[1] H. C. Claiborne, *Californium Production in the High Flux Isotope Reactor*, ORNL-CF-59-8-125 (August 1959).

[2] H. C. Claiborne, *Effect of Different Sets of Cross Sections on Cf^{252} Production in the HFIR*, ORNL-CF-59-10-19 (October 1959).

[3] H. C. Claiborne, *Effect of Non-Thermal Capture on Californium Production in the HFIR*, ORNL-CF-59-12-16 (December 1959).

[4] R. D. Cheverton, *HFIR Preliminary Physics Report*, ORNL-3006 (Oct. 4, 1960).

[5] E. P. Wigner, *Nucl. Sci. Eng.* 6(5), 420 (November 1959).

Table 7.3.1. General Nuclear Characteristics of the HFIR

Reactor power, Mw	100
Neutron flux, neutrons cm^{-2} sec^{-1}	
Maximum unperturbed thermal flux in island	5.5×10^{15}
Average thermal flux in island target (300 g of Pu^{242})	2.0×10^{15}
Average nonthermal flux in island target	2.4×10^{15}
Maximum nonthermal flux in fuel region	4.0×10^{15}
Maximum unperturbed thermal flux in beryllium reflector	
Beginning of fuel cycle	1.1×10^{15}
End of fuel cycle	1.6×10^{15}
Maximum unperturbed thermal flux at Be-H_2O reflector interface	
Beginning of fuel cycle	1.4×10^{14}
End of fuel cycle	1.7×10^{14}
Prompt-neutron lifetime, μsec	
Beginning of cycle	35
End of cycle	70
Effective delayed-neutron fraction	0.0071
Length of typical fuel cycle, days	23

There are basically three factors that control the magnitude of the island thermal-neutron flux. These are (1) the extent of neutron leakage from the fuel to the island, (2) the slowing-down and absorption characteristics of the island moderator, and (3) the diameter of the island. An optimum island diameter exists because of the conflicting requirements for complete moderation (large island) and for small volume (high neutron density). The use of an island moderator with the shortest slowing-down length results in the highest thermal-neutron flux because, for an equal number of neutrons moderated, the diameter (and thus volume) are the smallest. Even though light water has a relatively high thermal-neutron absorption cross section, its slowing-down length is short enough to result in a higher neutron flux in the island than that achieved with the other moderators considered.

The optimum island diameter selected for the HFIR, using light water as the island moderator, represents a compromise between achieving the maximum flux with and without the target in the island. Since the target displaces some of the island water, the optimum island diameter is dependent upon the plutonium loading (target size), the diameter being greater for larger targets. However, over the range of target loadings being considered (0–300 g of Pu^{242}), the calculated variations in optimum island diameter and corresponding maximum thermal-neutron fluxes were small. For this reason no attempt was made to optimize the island diameter for any specific target design.

Neutron leakage from the fuel to the island is a function of the core shape, the core size, the metal-to-water ratio, the power distribution, and the amount of parasitic absorption. Decreasing the core size and increasing the metal-to-water ratio increased the core leakage. Optimizing the length-to-diameter ratio of a given volume fuel region for a target length chosen equal to the active core length increased to a maximum the total leakage per unit core height into the island.

The limitation on reduction of the core volume was associated with heat-removal considerations. In this regard a very significant improvement in reactor performance was achieved by incorporating design features that resulted in a ratio of maximum-to-average power density of only 1.45 (exclusive of hot-spot factors). These features included cylindrical core geometry, a symmetrical reflector control system, extension of the aluminum ends of the fuel plates to suppress the neutron flux at the ends of the active fuel region, and radial variations in the fuel and burnable-poison concentrations.

As explained in greater detail in Sec. 5.2.2 and shown in Fig. 5.2.5(*a*), the radial variations in fuel and poison concentrations were achieved by varying the thicknesses of the fuel cores and burnable-poison cores across the width of the involute fuel plates. A disadvantage associated with this technique is that at the thin parts of the fuel-plate cores, the fuel and poison concentration inaccuracies due to segregation and inhomogeneity increase with increases in the ratio of nominal maximum-to-average fuel-plate core contour heights (decreasing thickness of the thin parts). To compensate to some extent for this trend, a 1-cm-thick water annulus was added between the two fuel annuli.[6] This helped to flatten the radial thermal-neutron-flux distribution and thus the fuel-plate core contours because thermal neutrons were added in the otherwise minimum thermal-neutron flux region, and because the associated permissible decrease in fuel loading decreased the radial thermal-neutron flux depression. This decrease in fuel loading had the further advantage of decreasing fuel segregation, which is a function of fuel density within the core. Although the addition of the water decreased by a few percent the island thermal-neutron flux per unit of reactor power, it appeared necessary in order to obtain reasonably accurate fuel and poison distributions.

To ensure that experimental facilities other than the island target did not impair the island neutron fluxes, all but a few rather small experimental facilities were relegated to a position in the beryllium reflector at least 3 in. from the fuel region. At this and further removed locations the effect of the beam holes and other anticipated experiments on power distribution was determined experimentally to be negligible; their total effect on reactivity was estimated to be less than 1%.

The radial thickness of the beryllium reflector was made somewhat greater than that needed for a reflector containing no experimental facilities. This was done to provide adequate high-neutron-flux space for the experimental facilities and also to help nullify their negative reactivity effect. The beryllium reflector thickness selected was approximately 30 cm.

7.3.2 Fuel and Burnable-Poison Loadings and Distributions

Detailed HFIR fuel cycle calculations[6,7] and critical experiment results[8] indicated that in order to achieve the desired average fuel cycle time of about 14 days, a fuel loading of 9.4 kg would be required. Enough boron burnable poison was added to achieve, in conjunction with the reflector control system, reasonable reactivity control. It was necessary that the resultant concentration of fuel and poison be consistent with fabrication and reactor performance considerations.

The radial distributions of the fuel and poison were established on the bases of providing the minimum and essentially constant value of peak-to-average power density during a fuel cycle and providing a satisfactorily high and constant thermal-neutron flux in the island. As shown in Fig. 5.2.5(*a*), the burnable poison was included only in the filler piece of the fuel-plate cores in the inner annulus; by doing this, the poison was added in the highest possible thermal-neutron flux areas consistent with the limitations imposed by fuel-plate core fabrication considerations. This resulted in the smallest poison loading for a desired amount of reactivity control, had the greatest effect on flattening of the thermal-neutron flux, and resulted in the greatest fractional burnup of the poison at the end of the cycle. Furthermore, the resultant overall neutron-flux depression in the inner annulus permitted shifting some fuel from the outer to the inner fuel annulus to help compensate for the very rapid burnup of fuel in the latter. All this helped to maintain a constant thermal-neutron flux in the island and a constant peak-to-average power density ratio in the fuel region; the absolute value of the latter ratio was also satisfactory.

---

[6]R. D. Cheverton, *Fuel-Cycle Analysis and Proposed Fuel and Burnable Poison Distribution and Loading for the HFIR and HFCE-2,* ORNL-CF-61-2-36 (February 1961).

[7]R. D. Cheverton, *HFIR Final Physics Report,* to be published.

[8]D. W. Magnuson (internal memorandum), *High Flux Isotope Reactor Critical Experiment No. 2* (September 1961).

7.3.3 Fuel Cycle Analysis[7]

Fuel cycle calculations were made to help determine the proper fuel and burnable-poison loadings and distributions and to investigate variations with time in such parameters as neutron flux and power distributions and reactivity. Figure 7.3.3 shows typical radial power distributions calculated for different times in the fuel cycle; Fig. 7.3.4 shows how the average thermal-neutron flux in a typical island target varies with time; and Fig. 7.2.2 shows, in addition to several clean-core radial flux distributions, a comparison between clean-core and end-of-cycle radial thermal-neutron flux distributions. As indicated by these curves, the island flux and the peak power density are a maximum at the beginning of the fuel cycle and decrease 2 or 3% during the fuel cycle.

The radial power distribution curves in Fig. 7.3.3 are typical for an average position along the length of the core. At the horizontal midplane the relative power density near the outer edge of the outer annulus is somewhat greater because of a symmetrical longitudinal flux peaking that takes place as the control plates are withdrawn from the core. Also, at the outer comers of the core the relative power density at the end of the fuel cycle tends to be greater than shown in Fig. 7.3.3. During a major portion of the fuel cycle, the thermal-neutron flux at these comer regions is depressed by the presence of the black regions of the control plates, resulting in a relatively low local percentage of fuel burnup. At the end of the fuel cycle, when the plates are fully withdrawn, both the fuel concentration and neutron flux are high, resulting in a somewhat higher power density than indicated. Additional calculations have been made which indicate that these longitudinal peaking effects will not produce relative peak power densities greater than those shown in Fig. 7.3.3.

Reactivity variations during the fuel cycle are shown in Fig. 7.3.5. As indicated by curves *A* and *D*, at the beginning of the fuel cycle the burnable poison controls about 0.04 Δk; at the end of a 14- to 15-day core life it is worth about 0.007 Δk, thus resulting in a loss of about two days of core lifetime. As previously mentioned, the actual fuel cycle is about 23 days, and therefore the burnable poison is worth less at the end of the cycle than indicated by these calculations.

The inclusion of a burnable poison in a reactor core often gives rise to a situation where the reactivity of the core will increase as some of the poison bums out. Curve *A* of Fig. 7.3.5 shows that this is not true in the case of HFIR operation. Curve *A*, however, does not represent the maximum possible reactivity. During the first seven days of full-power operation, the maximum k_{eff}, with the control plates out, occurs approximately four days after a shutdown from full power, at which time the combined poisoning effect of xenon and samarium is at a minimum. Curve *B* of Fig. 7.3.5 is a plot of k_{eff} achieved in this manner vs the time in the cycle at which the power was stepped from 100 to 0 Mw. Even in this case, k_{eff} is never greater than at the beginning of the fuel cycle. If, after being stepped from 100 to 0 Mw, the core is maintained down for ten days or more instead of just four days, the xenon is essentially gone and the promethium (Pm^{149}) has nearly all decayed to Sm^{149}, resulting in somewhat lower values of k_{eff} as depicted by curve *C*.

7.3.4 Temperature, Void, and Fuel Coefficients of Reactivity

Temperature, void, and fuel coefficients were determined experimentally,[9] except in a few cases where calculations[4,10] were used to predict the coefficients for core conditions that could not be adequately simulated in the critical experiments.

Temperature coefficients predicted for the clean-core condition are listed in Table 7.3.2.

At 108°F and without a target in the island, the predicted overall isothermal coefficient was 0.0; the change in *k* associated with an overall isothermal increase in temperature from 79°F to 108°F was $\sim +0.0007$.

[9] D. W. Magnuson (internal memorandum), *High Flux Isotope Reactor Critical Experiment No. 2, Part II* (January 1961).

[10] R. D. Cheverton, *Void Coefficient of Reactivity Associated with the Island Region of the HFIR*, ORNL-TM-114 (Nov. 15, 1961).

Table 7.3.2. Predicted Temperature Coefficients of Reactivity for the Clean Core

Condition	$(\Delta k/k)/°F$, Temperature Coefficient	
	At 79°F	At 155°F
With 310 g of PuO_2 target in island		
Overall isothermal	-1.16×10^{-5}	-3.1×10^{-5}
Fuel region only	-7.3×10^{-5}	-10.2×10^{-5}
Island + control + reflector regions (by difference)	$+6.1 \times 10^{-5}$	$+7.1 \times 10^{-5}$
Without target in island		
Overall isothermal	$+2.3 \times 10^{-5}$	-2.1×10^{-5}
Fuel region only	-6.0×10^{-5}	-11.7×10^{-5}
Island + control + reflector regions (by difference)	$+8.3 \times 10^{-5}$	$+9.6 \times 10^{-5}$

No experimental values for temperature coefficients were obtained for core conditions other than the clean condition. Analytical results indicate that as the fuel cycle proceeds, the core temperature coefficient becomes less negative; even so, the overall temperature coefficient with the target in the island remains negative for operation at any significant power level.

Void coefficients for the island with and without the target installed were also determined experimentally. As shown in Fig. 7.3.6, the void coefficients were positive up to void fractions of 70% without the target and 42% (based on the same total island volume) with a 300-g Pu target. The corresponding maximum changes in the neutron multiplication factor were +0.032 and +0.016 respectively. For void fractions less than about 20%, the void coefficients $(\Delta k/k)/(\Delta V/V)$ with and without the target appear to be about +0.05 and +0.06, respectively, in terms of the void fraction (V is total volume of the particular region). These island void effects were determined only for the clean-core condition. Variations with fuel cycle time are considered to be negligible.

Void coefficients were also determined experimentally for the fuel region. For the inner fuel, the central water, and the outer fuel annuli, the void coefficients in terms of the void fraction were −0.188, −0.046, and −0.384 respectively. As the fuel cycle progresses, these values will become somewhat less negative.

Fuel coefficients were determined experimentally for the clean-core condition. Average values for the inner and outer fuel annuli were 0.037 and 0.011 $\Delta k/k$ per kg of U^{235} respectively.

Measurements made in the reactor showed generally good agreement with the predicted values. Small differences were observed, due to changes in the control plate drive mechanisms, the water content of the control region, and differences between the fuel elements. Reports ORNL-CF-65-12-2 and ORNL-3573 contain additional information regarding the experiments in the reactor.

7.3.5 Reactivity Associated with the Plutonium Target[6]

The addition of an all-aluminum target to the island increases the neutron multiplication factor because of the positive void coefficient in the island. However, the net reactivity effect with the plutonium and subsequent transplutonium products included in the target is a function of time because the neutron cross section of the target varies with neutron dosage.[1] A target containing 1% of Pu^{241}, 1% Pu^{239}, and essentially 98% of Pu^{242} as feed material experiences maximum fission rates at the beginning of target irradiation and again about 0.4 yr later; at these times the absorption cross sections are about the same and so are the fission rates. At 0.1 yr the fission rate in the target is essentially zero (the Pu^{241} and Pu^{239} are nearly gone and the concentration of Cm^{245}, which is primarily responsible for the second fission rate peak, is not yet in significant abundance), and the absorption rate is a maximum. After 0.4 yr, at which time the second fission peak occurs, both the absorption and fission cross sections decrease (see Fig. 5.2.3).

Reactivities associated with the various target conditions are listed in Table 7.3.3.

Table 7.3.3. Reactivities Associated with an Island Target Initially Containing 310 g of $Pu^{242}O_2$ and 5.3 kg of Aluminum

Condition	Δk
Without target	0.000
With target	
No Pu^{241} or Pu^{239}, time zero	~0.000
1% Pu^{241} and 1% Pu^{239}, time zero	+0.008
0–1% Pu^{241} and 0–1% Pu^{239}, 0.1 yr (min. Δk)	−0.001
0–1% Pu^{241} and 0–1% Pu^{239}, 0.4 yr (max. Δk)	+0.008

7.3.6 Nuclear Characteristics of Control Plate

The control system used in the HFIR was selected primarily for its ability to adequately control reactivity without introducing undesirable perturbations and asymmetries in the fuel element power distribution and neutron fluxes in the target. Basically, the system constitutes a reflector control system that regulates the flow of thermal and epithermal neutrons from the beryllium reflector to the fuel region. As shown in Fig. 7.2.1, the narrow annulus between the fuel region and the beryllium reflector effectively contains two, thin, concentric cylinders that are separated from each other and from the adjacent regions by narrow coolant gaps, as described in Sec. 5.2.4. The inner cylinder has a single drive rod and is used for both shim and regulation. The outer cylinder is divided into quadrants, each with its own drive rod and release mechanism; these four control plates are used for both shim and safety. During normal operation, the four shim-safety plates are moved in concert so as to minimize asymmetries in the power distribution; however, when used in emergency, they are released separately and thus provide multiplicity.

In order to maintain the longitudinal power distribution variations within acceptable limits and in order to prolong the neutron-absorption life of the plates, the control cylinders were divided, with respect to neutron-absorption capability, into three discrete longitudinal regions: a highly neutron-absorbing (black) region, a moderately neutron-absorbing (gray) region, and a comparatively poor neutron-absorbing (white) region. By locating the black regions of the two control cylinders at opposite ends of the core, as shown in Fig. 7.2.1, and by moving the cylinders in opposite directions in a symmetrical fashion, as shown in Fig. 5.2.9, it is possible to maintain power distribution symmetry about the horizontal midplane of the core.

Materials for the control plates (Fig. 5.2.8) are a ~31-vol % Eu_2O_3-Al dispersion clad with aluminum for the black region, a ~38-vol % tantalum-aluminum dispersion clad with aluminum for the gray regions, and solid aluminum for the white regions. In view of the significantly larger absorption cross sections of aluminum and water relative to that for the beryllium reflector, the radial thicknesses of the three control region coolant channels and of the control plates were minimized in a manner consistent with mechanical and hydraulic considerations, in order to minimize the negative reactivity effect of the control region at the end of a fuel cycle.

The integral and differential reactivity worths for the control plates were determined experimentally[11] using a set of Eu_2O_3-Ta-Al plates that are essentially identical to the reactor grade plates. With all plates fully inserted, the total control plate worth was 0.191 Δk. With the inner cylinder fully withdrawn, the four shim-safety plates had a total worth of 0.165 Δk. The corresponding values for a similar set of control plates in the HFIR are 0.187 Δk for all plates inserted and 0.147 Δk for the four shim-safety plates alone. The difference in control plate worth is associated with a difference in water content and distribution in the control region of the critical experiment and the HFIR facilities. Shutdown margins for various core and control plate conditions are shown in Table 7.3.4 and are discussed more fully in ORNL-3573.

Differential worths (Δk per unit distance of control plate travel) were determined for the four shim-safety plates with the four plates effectively in concert and with the inner cylinder held stationary. The results of these tests are shown in Fig. 7.3.7. Differential worths of the shim-regulating cylinder are somewhat greater than those for the shim-safety plates. Not shown in Fig. 7.3.7 are the differential worths of the control plates for some extreme asymmetrical positions of the inner control plate relative to the outer plate. Under some of these conditions, which can occur during mode 2 and 3 operation, the differential worth of the shim-regulating cylinder can be as large as 0.0142 Δk/in. and that for the combined safeties as large as 0.0135 Δk/in. These two maximum conditions cannot occur simultaneously. When one is a maximum, the other is a minimum. The largest combined differential worth occurs with the control plates symmetrical and is equal to about 0.021 Δk/in.

[11] D. W. Magnuson (internal memorandum), *High Flux Isotope Reactor Critical Experiment No. 2, Part VIII* (October 1962).

Table 7.3.4. Shutdown Margins for the 9.4-kg HFIR Core (with 2.8 g B^{10})

	Shutdown Margin (Δk)	
	5	4[a]
1. Maximum reactivity case, exclusive of accidental reactivity additions, with a new fuel element and the following conditions: (*a*) Extreme permissible loadings and distributions of fuel and burnable poisons resulting in maximum positive reactivity (*b*) Beryllium not poisoned (c) Maximum target reactivity (*d*) Zero power at 70°F	0.069	0.030
2. Typical nominal case, with a new fuel element and the following conditions: (*a*) Nominal loadings and distribution of fuel and burnable poison (*b*,*c*, and *d*) as in case 1	0.081	0.042

[a] Four shim-safety plates in, shim-regulating cylinder out.

7.3.7 Summary of Reactivity Accountability

Pertinent reactivities associated with the 9.4-kg HFIR core are summarized in Table 7.3.5.

Table 7.3.5. Summary of Reactivity Accountability for the 9.4-kg Core

Parameter	Reactivity (Δk)		
	Time in Cycle (days)		
	0	2	14
Fuel worth with following core conditions: no boron burnable poison; no target; no Be poisoning; zero power at 70°F	0.135		
Boron burnable poison (2.8 g of B^{10})	–0.05	–0.037	–0.009
Temperature deficit (evaluated with a 310-g PuO_2 target): zero power at 70°F to 100 Mw	–0.004		
Plutonium target: 310 g $Pu^{242}O_2$ + 3 g $Pu^{241}O_2$ + 3 g of $Pu^{239}O_2$			
Maximum (time zero and again at 0.4 yr)	+0.008		
Minimum (0.1 yr)	0.001		
Xe^{135} + Sm^{149} (at power)	0	–0.049	–0.053
All fission products	0	–0.053	–0.086
Be poison[b] (Li^6 + He^3)			
Time zero	0		
0.2 yr	–0.013		
5 yr	–0.016		
Beam tube flooding	~0		
Fuel loading tolerance (±1%)	±0.0015		
Boron loading tolerance (±10%)	±0.0038		
Fuel distribution tolerance (±10%)	±0.0054		
Boron distribution tolerance (±35%)	±0.0023		
Minimum $k_{eff}-1$ (clean core, 100 Mw)	0.054		
Typical nominal[a] $k_{eff}-1$ (clean core, 70°F)	0.093		
Maximum $k_{eff}-1$ (clean core, 70°F)	0.106		
Shutdown margins for typical nominal[a] k_{eff} case:			
All plates inserted	0.081		
Inner cylinder withdrawn, 4 plates inserted	0.042		

[a]See Table 7.3.4 for description of "typical nominal" core.

[b]No significant poisoning effect which could be attributed to the beryllium reflector was observed at the end of ~300 full-power days of operation.

7.3.8 Method of Nuclear Analysis

Both analytical and experimental analyses have been used to arrive at the final core design. Analytical methods included 33-group, one-dimensional, diffusion theory;[12] four-group, one-dimensional, transport theory;[13] and two- and four-group, two-dimensional, diffusion theory.[14] The diffusion theory reactor codes were used more extensively because they provided the only practical methods for analyzing the fuel cycle in considerable detail; and for this purpose they were satisfactory. In order to simulate the radial distribution of fuel and burnable poison, the fuel region was usually divided into 17 discrete radial regions.

Neutron cross sections for the HFIR were calculated using a multiregion thermalization transport theory code[15] for the thermal neutrons and a single-region, multigroup, consistent P_1 theory code[16] for the nonthermal neutrons. The use of the thermalization code made it possible to obtain spectrum-averaged thermal-neutron cross sections as a function of radial position in the fuel region, thus accounting for spectral hardening as the thermal neutrons return from the island, water annulus, and beryllium reflector and penetrate the fuel region.

Four HFIR critical experiments have been conducted, three in a critical facility and the fourth in the reactor. In addition, the results from Russian flux-trap critical experiments[17] were used to verify our method for analytically determining the optimum diameter for the island.

In the first critical experiment,[18–20] a solution fuel was used in a single annular fuel region. A mixture of H_2O and D_2O was used as the diluent to simulate the nuclear characteristics of the actual H_2O-Al lattice. Heavy water was also used as the side reflector in lieu of beryllium. With the exception of core height, the core dimensions were the same. Critical mass, axial buckling, power distribution, and island void coefficients were the main parameters investigated. Most of these data were used to help establish the adequacy of the analytical techniques.

The second critical experiment[8,9,21–28] more nearly resembled the actual HFIR core. The fuel elements were essentially the same as described herein, with the exception that the U^{235}

---

[12] J. Replogle, *Modric: A One-Dimensional Neutron Diffusion Code for the IBM-7090*, K-1520 (Sept. 6, 1962).

[13] B. G. Carlson, *The S_n Method and the SNG Code*, LAMS-2201 (1959).

[14] M. L. Tobias and T. B. Fowler, *The Twenty-Grand Program for the Numerical Solution of Few-Group Neutron Diffusion Equations in Two Dimensions*, ORNL-3200 (Feb. 7, 1962).

[15] H. C. Honeck, *Thermos: A Thermalization Transport Theory Code for Reactor Lattice Calculations*, BNL-5826 (September 1961).

[16] G. D. Joanou and J. S. Dudek, *GAM-1: A Consistent P_1 Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants*, GA-1850 (June 28, 1961).

[17] S. M. Feinberg *et al.*, *Proc. Intern. Conf. Peaceful Uses At. Energy, 2nd Geneva*, **10**, 296 (1958).

[18] H. C. Claiborne, *Solution Critical Experiments for the HFIR: Preliminary Calculations*, ORNL-CF-61-8-37 (August 1961).

[19] J. K. Fox, L. W. Gilley, and D. W. Magnuson, *Neutron Phys. Div. Progr. Rept. Sept. 1, 1960*, ORNL-3016, p. 59.

[20] J. K. Fox, L. W. Gilley, and D. W. Magnuson, *Preliminary Solution Critical Experiments for the High-Flux Isotope Reactor*, ORNL-3359 (May 28, 1963).

[21] D. W. Magnuson, *The U^{235} Content of Foils Punched from HFIR Critical Experiment No. 2 Fuel Plates*, ORNL-CF-61-7-50 (July 1961).

[22] D. W. Magnuson, *High Flux Isotope Reactor Critical Experiment No. 2, Part III*, ORNL-CF-62-1-53 (January 1962).

[23] D. W. Magnuson, *High Flux Isotope Reactor Critical Experiment No. 2, Part IV*, ORNL-CF-62-5-20 (May 1962).

[24] D. W. Magnuson, *High Flux Isotope Reactor Critical Experiment No. 2, Part V*, ORNL-CF-62-5-10 (May 1962).

[25] D. W. Magnuson, *High Flux Isotope Reactor Critical Experiment No. 2, Part VI*, ORNL-CF-62-8-25 (August 1962).

[26] D. W. Magnuson, *High Flux Isotope Reactor Critical Experiment No. 2, Part VII*, ORNL-CF-62-9-75, (September 1962).

[27] P. R. Kasten and R. D. Cheverton, *Revised Version of HFIRCE-2*, ORNL-CF-61-1-42 (January 1961).

[28] D. W. Magnuson and J. K. Fox, *Neutron Phys. Div. Ann. Progr. Rept. Sept. 1, 1961*, ORNL-3193.

loading was 8.01 instead of 9.4 kg, and the boron burnable poison was distributed uniformly in both the filler and fuel section of the inner element fuel-plate core. The control plates were different only in absorber material and length of the gray region.

The following parameters were investigated in the second critical experiment:

Power distribution	Void coefficients of reactivity
Flux distribution (thermal)	Fuel coefficients of reactivity
Control plate worth	Plutonium target worth
Control plate differential worth (Δk/in.)	Reactivity with fuel elements submerged in H_2O
Fuel worth	Neutron lifetime
Temperature coefficients of reactivity	Beam tube flooding

Power distributions were obtained by counting punchings from the fuel-bearing portion of removable fuel plates. Most of the changes in the neutron multiplication factor were determined by means of the pulsed neutron technique.

With the aid of these data, improvements were made in the calculational methods, the fuel loading was increased from 8.01 to 9.40 kg, and the design of the control plates was modified to increase their differential worth (Δk/in.).

The third critical experiment[29,30] incorporated all the modifications associated with the 9.4-kg core, including the Eu_2O_3-Ta-Al control plates, and was used primarily to check the power distribution, the control plate integral and differential worth, and the fuel worth; it was also used to investigate the reactivity variations associated with fuel element handling.

A fourth critical experiment[31] was conducted in the actual HFIR facility. Many of the above experiments were repeated for check purposes.

7.4 Fuel Element Design and Analysis

7.4.1 General Design Considerations

(a) General Criteria. – The detailed design of the HFIR fuel elements was of necessity strongly influenced by the primary objective of the reactor. As discussed in Sec. 7.3.1, a very high power density and thus small core volume and high heat transfer surface-to-volume ratio are required to achieve a high thermal-neutron flux in the island per unit of reactor power. To help achieve the desired small core volume, the ratio of maximum-to-average power density was made as small as possible. This was done in part by specifying a cylindrical core geometry and a radial variation in the fuel and burnable-poison distribution, both of which provided additional criteria for the fuel elements.

Also on the basis of nuclear considerations a specific length-to-diameter ratio of the core and a high and uniform metal-to-water ratio ($\sim$1.0) were established as important general criteria for the fuel elements. Additional criteria included the use of aluminum for cladding material, a proposed 15-day duration at 100 Mw, and, of course, an economically reasonable design.

(b) Selection of Fuel Element Type. – After analyzing several types of fuel elements that could possibly satisfy the above criteria, a cylindrical, annular type containing involute geometry fuel plates was selected. The thickness of the fuel core in each fuel plate was varied across the width of the fuel plate [as shown in Fig. 5.2.5 (*a*)] to provide the desired radial fuel and burnable-poison distribution, and the cylindrical geometry was in keeping with the specified cylindrical shape of the core. As shown in Fig. 5.2.4, two such elements, containing slightly different fuel plates, were used to make up a core loading.

The chosen element adequately satisfied the general criteria and had three further important advantages: (1) the involute plate geometry has desirable characteristics in connection with

[29] R. D. Cheverton (internal memorandum), *HFIR Critical Experiment No. 3 (HFIRCE-3)* (December 1962).

[30] See also reports ORNL-CF-64-7-74; ORNL-CF-65-2-64; ORNL-CF-66-4-10; ORNL-CF-66-8-31; and ORNL-TM-1488.

[31] ORNL-CF-65-12-2.

hydraulically and thermally induced loads and deflections; (2) only two types of fuel plates, both involute geometry but differing in dimensions and fuel loading, are required; and (3) preliminary fabrication development work on involute geometry and variable-thickness fuel cores indicated feasibility of construction.

(c) Basic Design Problems and General Method of Analysis. – The high performance demand on the fuel elements required high coolant velocities, narrow coolant channels, thin fuel plates, and high temperatures. This combination introduced problems associated with corrosion, with hydraulically and thermally induced loads and deflections, and with fabrication tolerances; thus, a very thorough analysis of the fuel element was required. In order to minimize many of the uncertainties encountered in the analysis, an extensive experimental program was conducted. Some of the more important experiments were related to the following specific subjects:

1. static and long-term (creep) deflection characteristics of involute geometry fuel plates subjected to differential pressures;
2. longitudinal, thermally induced, static and creep buckling characteristics of involute geometry fuel plates;
3. fuel-plate material physical properties;
4. fuel-plate corrosion rates and thermal resistance of adherent corrosion product films;
5. burnout heat flux, coolant film heat transfer coefficient, and friction factors specifically for HFIR conditions;
6. power distribution in core;
7. fabrication and assembly tolerances associated with HFIR fuel elements.

The analysis of the fuel elements was performed in such a way that the mechanical, hydraulic, and thermal parameters were considered simultaneously. This approach facilitated achieving the maximum possible performance consistent with HFIR operating conditions. Insofar as steady state operating conditions are concerned the power level at which hot spot incipient boiling occurs was considered rather than the power level at which burnout occurs. The reason for this is that near the burnout power level the calculated coolant channel gaps became quite narrow, thus making the applicability of available burnout correlations open to question. In some extreme cases of channel narrowing there is an indication from a few tests that the burnout heat flux tends to approach the incipient boiling heat flux. For very low power and low flow operation, such as for Mode 3, the burnout data were applicable and therefore were used. Details of the analysis are discussed in Secs. 7.4.2 and 7.4.3.

(d) Summary of Pertinent Design Data for the HFIR Fuel Elements. – Descriptive and heat transfer data related to the final design of the fuel elements are listed in Table 7.4.1.

7.4.2 Mechanical and Hydraulic Analysis[32–35]

The fuel plates in the HFIR are subjected primarily to two types of loading: (1) that resulting from hydraulically induced, lateral pressure differentials across the fuel plates and (2) that resulting from thermally induced, differential radial and longitudinal expansions between adjacent fuel plates and side plates.

The hydraulically induced loads exist because small dimensional differences cause velocity and pressure differences between adjacent coolant channels. This phenomenon was integrated

---

[32] T. G. Chapman, *Thermal Expansion and Pressure Differential Induced Stresses and Deflections in HFIR Involute Contoured Fuel Plates*, to be published.

[33] J. R. McWherter, T. G. Chapman, and C. A. Burschted, *Deflection of a HFIR Involute Fuel Plate Under a Uniform Load*, to be published.

[34] J. R. McWherter and T. G. Chapman, "Mechanical and Hydraulic Design of the HFIR," p. 99 in *Research Reactor Fuel Element Conference*, TID-7642 (Book 1).

[35] Cheverton and Kelly, *Deflection Characteristics of a HFIR Fuel Plate*, to be published.

Table 7.4.1. Summary of Pertinent Design Data for the HFIR Fuel Elements

Type of core	Cylindrical annulus, flux trap
Type of fuel elements	Cylindrical annuli (2); aluminum fuel plates in involute geometry
Fuel-plate thickness, in.	0.050
Coolant-channel thickness, in.	0.050
Length of fuel plates, in.	24
Length of active core, in.	20
Inside diameter, inner fuel element, in.	5.067
Outside diameter, inner fuel element, in.	10.590
Inside diameter, outer fuel element, in.	11.250
Outside diameter, outer fuel element, in.	17.134
Active volume, liters	50.59
Heat transfer surface area, ft^2	428.8
Fuel element materials	
Side plates	Aluminum
Cladding and core matrix	Aluminum
Fuel	U_3O_8(93% U^{235}) dispersed in aluminum
Fuel loading, kg of U^{235}	9.40
Inner element	2.595
Outer element	6.805
Burnable poison	B_4C dispersed in aluminum
Burnable poison loading (inner element), grams of B^{10}	2.8
Heat removal	
Total reactor power, Mw	100
Power density (97.5 Mw in fuel region), Mw/liter	
Average	1.93
Maximum	4.384
Heat flux (97.5 Mw conducted to coolant), Btu hr^{-1} ft^{-2}	
Average	7.76×10^5
Hot spot	1.97×10^6
Incipient boiling power level at steady state, Mw	
Beginning of fuel cycle	
600-psi core inlet pressure	142
900-psi core inlet pressure	165
End of 25-day fuel cycle	
600-psi core inlet pressure	157
900-psi core inlet pressure	180
Coolant velocity (average), fps	51

into the overall fuel element analysis by developing analytical expressions for the local velocity distribution, velocity pressure drop, and frictional pressure drop in channels distorted by gradual deviations such as those shown in Fig. 7.4.1. From these equations it was possible to calculate the lateral pressure differentials across a fuel plate.

Deflections and stresses resulting from the above pressure differentials were studied both analytically and experimentally for the involute geometry fuel plate. Results of these studies indicated that when the involute was subjected to a uniformly distributed load, the deflected shape was in the form of an S, relative to the initial involute contour; that is, both positive and negative deflections existed, as shown in Fig. 7.4.2. The advantage of this condition is that the

Table 7.4.1. (continued)

Temperatures, °F (with 121°F inlet water temperature)

	Start of Cycle			End of 25-day Cycle		
	Fuel Nominal	Hot-Streak	Hot Spot	Fuel Nominal	Hot-Streak	Hot Spot
Fuel plate temperature, °F	158–287	351	426	153–325	455	620
Metal-oxide interface temperature, °F	152–269	329	387	149–303	436	589
Oxide-water interface temperature, °F	152–269	329	387	146–257	295	343
Water outlet temperature, °F	164–196	249	249	163–188	227	227

total cross-sectional flow area of a channel tends to remain constant as adjacent plates deflect; this reduces further changes in the lateral pressure differentials and thus reduces the plate deflection. A further advantage of the above involute-curve deflection characteristic is that the critical velocity tends to be quite high as compared with a uniformly curved plate. The estimated critical velocity for buckling-type failure for the HFIR fuel elements is well above the maximum operating coolant velocity.

Thermally induced deflections of the plates were also studied both analytically and experimentally. Although both radial and longitudinal expansions exist, the longitudinal expansions are by far the more significant. In this case the important temperature difference is that between the fuel plate and the two side plates. Since the fuel plate is hottest, it is in longitudinal compression and tends to buckle. Initially the results of analyses and preliminary tests indicated that the buckling would be in the form of a sine wave with a length of about 2 in., as shown in Fig. 7.4.1(a). This model was used in the early thermal-hydraulic analyses.[39] Later, a detailed experimental program was conducted on fuel plate buckling which showed that the plates actually do not buckle in the form of multiple sine waves but rather deflect in the shape of a single wave, as shown in Fig. 7.4.1(b), with all plates deflecting in the same direction. Furthermore the experiments showed that much larger axial differential expansion than anticipated would be required to produce the multiple-wave curve, thus indicating a substantial margin of safety. These later data were incorporated into the overall heat transfer analyses as set forth in Ref. 40.

In the above pressure and thermal deflection tests the creep behavior of the fuel plates at appropriate temperatures was also studied, and the results were incorporated into the fuel element analysis.

7.5 Heat Transfer Design[36–40]

7.5.1 Heat-Removal Analysis Method

A more-or-less conventional approach to the heat-removal analysis was adopted. It was assumed, in order to guarantee an adequate burnout margin in spite of the lack of sufficient statistical data on many of the factors involved, that all hot-spot factors would be simultaneously superimposed in the worst possible way at one given point in the reactor. The principal disadvantage associated with this approach is that it can result in an unnecessarily high degree

[36] N. Hilvety, *Preliminary Hot Spot Analysis of the HFIR*, ORNL-CF-60-3-12 (March 1960).

[37] N. Hilvety, *Preliminary Hot Spot Analysis of the HFIR*, ORNL-CF-60-3-12 suppl. (March 1960).

[38] N. Hilvety and T. G. Chapman, *Summary of HFIR Hot Spot Studies*, ORNL-CF-62-1-52 (January 1962).

[39] N. Hilvety and T. G. Chapman, *HFIR Fuel Element Steady-State Heat Transfer Analysis*, ORNL-TM-1903 (December 1967).

[40] H. A. McLain, HFIR *Fuel Element Steady-State Heat Transfer Analysis, Revised*, ORNL-TM-1904 (December 1967).

of conservatism, thus imposing a significant limitation on reactor performance. For this reason considerable emphasis was placed on understanding and minimizing the uncertainty and tolerance factors that are used in such an analysis.

There were five basic types of factors considered in this analysis (the individual factors and typical values are given in Table 7.5.1):

1. neutron-flux variations;
2. fuel element fabrication tolerances (coolant channel dimensions, plate fuel content, fuel distribution, etc.);
3. fuel-plate defects (nonbonds and local fuel segregation);
4. changes in coolant channel dimensions caused by fuel-plate deflections resulting from hydraulically and thermally induced loads;
5. uncertainties in plant-operating parameters (neutron-flux peaking, power level, coolant temperature, pressure level, circulation rates, etc.).

The heat transfer system was mathematically so described that each of the heat transfer parameters, including flow rates, coolant temperature rises, film temperature drops, etc., was a function of the tolerances and the uncertainty and defect factors by which it could be influenced. In this way the effect of an individual factor could be conveniently determined, and the relatively crude method of simply adding approximate factors together to obtain an "overall" hot-spot factor

Table 7.5.1. Principal Factors Used in the Heat-Removal Analysis[40]

	Typical Value
Fuel Element Fabrication Tolerances	
Average coolant channel thickness variation, mils	6
Maximum coolant channel thickness variation, mils	10
Deviation in total plate fuel loading, percent of design values	101
Deviation in plate fuel loading across radial profile, percent of design value	110
Deviation in local fuel plate loading, percent of design value	130
Fuel Plate Defects	
Local heat-flux peaking caused by nonbonds, fractional increase	1.16
Local heat-flux peaking caused by fuel segregation, fractional increase	1.30
Uncertainties in Performance and Operating Conditions	
Reactor power level	1.02
Total heat transfer area	1.045
Power density distribution	1.10
Inlet coolant temperature	1.01
Friction factor	1.05
Local heat transfer correlation	0.90
Burnout correlation	0.80
Incipient boiling correlation	1.00
Hot streak factor	1.10
Flux peaking for fuel extending beyond normal boundaries	1.23

was avoided. This method also eliminated inclusion of conflicting factors in the same calculation and gave proper advantage-credit to one parameter caused by a disadvantage to another. For example, the same calculation cannot use the assumption of adjacent wide and narrow channels for calculating pressure-induced fuel-plate deflection while simultaneously assuming adjacent narrow channels for calculating plate temperature and buckling; likewise, factors that cause an increase in bulk water temperature also result in an increased heat transfer coefficient. In determining the incipient boiling point, all temperature-dependent variables were evaluated at the actual incipient boiling power level – rather than at the normal operating power, as done in more conventional analyses. This had the advantage of establishing the maximum steady-state power level (with a constant inlet temperature of 120°F), rather than the less restrictive case of the increase in local heat flux which would cause incipient boiling or burnout.

In analyzing the core, four conditions were considered: (1) the average of the core; (2) the hot and cold plates which include all the factors which affect significantly large areas of the plate (from the standpoint of plate structural integrity and deflections); (3) the hot streak, which extends the entire length of the core and influences bulk water temperature at the hot spot but does not affect plate structural integrity or deflection; and (4) the hot spot, which is very local in nature, provides no feedback, and is the point at which incipient boiling is assumed to occur.

Experimental data correlations were examined to ensure that they applied at HFIR operating conditions and that the experimental data spread was reduced to the practical minimum. Special experiments, as noted in Sec. 7.5.2, were performed to simulate as nearly as feasible the actual HFIR conditions.

Many factors not normally considered in heat-removal studies were included directly in the HFIR analysis. Local heat-flux peaking factors, such as nonbonds and fuel segregations, were considered in addition to the effects of fuel-plate deflections. In the case of those factors which could be limited to small plate areas by proper inspection techniques, both heat conduction in the plane of the plate and the effects of changes in the local heat transfer coefficient on this conduction were taken into account. In considering the increase in local heat flux necessary to reach burnout it was assumed in evaluating these factors that the burnout heat flux correlation was equally applicable to large or small areas. The buildup of oxide film on the fuel plates,[41] which enters into the fuel-plate deflections by affecting both the fuel-plate mechanical strength and the temperature differences that cause thermal deflections was considered.

In evaluating the factors that give rise to hot spots, it was necessary to consider hot and cold plates and hot streaks in addition to hot spots. Since plate deflections are based on the average plate temperature at the hot-spot elevation, anything that can influence this average temperature must be applied as hot and cold plate factors. The tolerance on the average coolant channel thickness, which determines the channel flow area, is such a factor since it affects the entire channel width. The tolerance on the local coolant channel thickness, which influences only a small portion of the channel, can, at worst, influence a narrow streak down the plate length; so it was applied as a hot-streak factor. Plate loading tolerances can affect an entire plate, while a tolerance on fuel contour is influential only over a local spot or, at worst, a hot streak. Local heat flux peaking caused by nonbonds and segregation can be limited to small plate areas by proper inspection techniques, and therefore was considered as a hot-spot factor.

The basic approach to the HFIR hot-spot analysis of assuming all factors to be simultaneously superimposed at a given point is conventional. However, the application of this approach and the degree of detail involved in this particular study depart from, and is believed to be an improvement to, the conventional.

7.5.2 Experimental Program

The experimental program accompanying the design of the HFIR fuel element may be summarized under the general headings of fuel-plate deflection experiments, hydraulic tests, heat

[41] J. C. Griess *et al.*, *Effect of Heat Flux on the Corrosion of Aluminum by Water. Pt. III. Final Report on Tests Relative to the High Flux Isotope Reactor*, ORNL-3230 (Dec. 25, 1961).

transfer and burnout experiments, corrosion tests, critical experiments, and fabrication development. The hydraulic experiments consisted of flow calibration tests on the labyrinth seal between fuel annuli, flow tests on individual fuel annuli assemblies, and reactor hydraulic mockup tests which incorporated a complete core assembly. Individual annuli were flow tested in order to examine structural integrity and to establish average fuel element friction drop correlations. Mockup experiments included tests on circumferential flow variation and total fuel element flow rate. Labyrinth seal calibration tests were conducted in order to establish the length of seal area required to control the flow in the annulus between fuel assemblies.

Heat transfer and burnout tests[42] were run in order to establish the correlations and uncertainty factors to be applied in the hot-spot analysis. Corrosion experiments[41] established the feasibility of 6061 aluminum as a fuel element construction material and furnished oxide-film buildup correlations for use in calculating fuel-plate temperatures.

Another important experimental program in connection with the HFIR fuel element design was that concerned with fabrication and inspection of the fuel elements.[43–45] The required radial fuel distribution with its associated tolerances required an extensive development effort, both from the standpoint of fabrication and of inspection. The plate-forming and assembly procedures also required considerable development work in order to ensure that such complex elements could be assembled to the specified tolerances without allowing large contingencies for rejects.

7.5.3 Results of Heat-Removal Analyses

Some of the results of the HFIR heat-removal calculations are presented in Table 7.4.1. Heat flux and operating-temperature values are listed for both beginning and end of cycle for the average fuel element conditions, the outlet from the maximum hot streak, and the maximum hot spot. The incipient boiling power level increases during the cycle due to redistribution of the power density (from 142 to 157 Mw). The oxide buildup results in an appreciable increase in plate temperatures, as evidenced by the hot-spot metal temperature, which increases by about 200°F during the fuel cycle. Since plate strength is dependent upon the hot-plate temperatures (which are somewhat lower than the hot-streak values listed here), it appears from these data that, with the possible exception of very local hot spots, the temperatures are such that plate strengths can be expected to remain at reasonable values throughout the fuel cycle.

The effects of vessel inlet pressure and coolant flow rate were of special interest in establishing the desired reactor operating conditions and in examining effects such as loss of coolant flow (Fig. 7.5.1). At low inlet pressures the incipient boiling power level is highest for an intermediate flow rate because at higher flows the pressure drop across the core results in a lowered pressure at the hot spot. It is for this reason that power to the main coolant pump motors is, during a loss-of-pressure incident, shut off after the reactor is scrammed (see Sec. 8.8).

The typical effect of coolant-channel tolerances on minimum calculated incipient boiling power level is shown in Fig. 7.5.2. These curves indicate that for a constant average channel thickness tolerance, a change in local channel thickness tolerance of 5 mils results in about a 9% change in incipient boiling power level. A 5-mil change in average channel thickness tolerance, with the local tolerance constant, would result in about a 11% change in incipient boiling power level. Studies such as this have been used to establish the required fuel element fabrication tolerances.

---

[42] W. R. Gambill and R. D. Bundy, *HFIR Heat Transfer Studies of Turbulent Water Flow in Thin Rectangular Channels*, ORNL-3079 (June 5, 1961).

[43] M. M. Martin, J. H. Erwin, and C. F. Leitten, Jr., "Fabrication Development of the Involute-Shaped High Flux Isotope Reactor Fuel Plates," p. 268 in *Research Reactor Fuel Element Conference*, TID-7642 (Book 1).

[44] J. W. Tackett *et al.*, "Assembly and Welding Development for the High Flux Isotope Reactor Fuel Element," p. 290 in *Research Reactor Fuel Element Conference*, TID-7642 (Book 1).

[45] R. W. McClung, "Nondestructive Testing of High Flux Isotope Reactor and Advanced Test Reactor Fuel Elements," p. 337 in *Research Reactor Fuel Element Conference*, TID-7642 (Book1).

The results of the fuel element design analysis indicated that a fuel element constructed to the specifications employed in the heat-removal calculations should be capable of meeting the overall HFIR performance specifications. Operation of the HFIR for about one year has shown the design to be satisfactory at the normal full power of 100 Mw.

7.5.4 Transient Hot-Spot Studies

In addition to the steady-state hot-spot data, useful information was obtained from the heat-removal analysis for conducting transient studies. The time constant for transfer of heat from the fuel plates to the coolant in the HFIR is so small that relatively slow transients, such as pump coastdown, are essentially equivalent to steady-state operation so that the steady-state procedure can be used directly. For extremely fast transients, the steady-state results give a possible basis for estimating the consequences of the excursion. It may be conservatively assumed that, during a transient, film blanketing occurs when the hot-spot heat flux reaches the burnout or incipient boiling heat flux calculated by the steady-state procedure. The safety system can then be required to limit the peak heat flux to this value, which would probably result in no permanent core damage; or it can be assumed that the hot spot becomes insulated at this heat flux, and hot-spot melting can be set as the limiting level for the excursion. An excursion of this magnitude would result in permanent core damage and would necessitate replacement of the fuel element, but it would not be expected to result in excessive contamination of the reactor system due to the small area affected.

HFIR transients resulting from reactivity accidents have been studied in some detail[46,47] by means of analog computer simulation. The results indicate that the reactor can survive all reasonable reactivity accidents without any melting of the core. See also Sec. 8.8.3.

---

[46] ORNL-TM-1747.

[47] ORNL-3573.

8. INSTRUMENTATION AND CONTROL

8.1 Introduction

The HFIR safety and control systems have been designed to provide for safe and orderly operation of the reactor from a central control room. Both of these systems make extensive use of process information such as coolant flow, temperature, and pressure; thus these parameters, together with others not directly related to control of the reactor power, are also regulated from the central control room. Essentially all routine operations, including startup and shutdown, can be monitored and controlled from this location.

The control system is designed to relieve the operator of routine manipulations by enabling the instrument which senses a change in the system parameters to also initiate the required corrective action. This approach is consistent with the philosophy of utilizing the operator to supervise the functions of the control system rather than to include him as an integral part of it. Nevertheless, certain actions will still be required of the operator. In particular, any increase in reactivity beyond that allotted to the automatic power regulation feature 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, it is not anticipated that the safety system will often be called upon; however, it is independent of the control system and capable of very fast response when needed.

At power levels above 10 Mw, control of the reactor is based primarily upon measurement of the heat-power level. This is accomplished by detecting the rate of coolant flow and its temperature rise as it passes through the core. These two signals are combined in heat-power calculators which compute and display the actual heat-power level. Although the heat-power instrumentation is quite accurate, its time response is too slow to permit its use as the sole source of safety or control action. These actions are directly initiated by signals from neutron-detecting devices which are continually and automatically calibrated using the output of the heat-power calculators. This arrangement permits fast action by the nuclear instrumentation and, at the same time, allows a high degree of accuracy regardless of the operating history of the core, the position of the control plates or ionization chambers, or any other effects which may cause the nuclear signal to vary in a manner not proportional to the power.

Signals from the flow-measuring device are also used to ensure that the control system will automatically limit the maximum allowable operating power to a level consistent with the actual flow rate and to automatically set the neutron flux trip levels in the safety channels at appropriate values. The necessity for this action is based upon the design requirements that for any flow rate the reactor operate normally in the flow-power region well below that point at which the onset of nucleate boiling occurs, and that any abnormal power increases be terminated before the hot-spot incipient boiling point is reached. The normal operating range and the locus of the values at which safety action occurs are shown in Fig. 8.1.1.

The reactor inlet-temperature and -pressure signals are incorporated separately into the safety systems as "on-off" signals since little variation is anticipated during normal operations. Should the inlet temperature exceed, or the inlet pressure become less than, certain preset levels, a reactor scram will result.

The HFIR control and safety systems are unique at ORNL in that in each system three channels of instrumentation are used, with the concurrence of two of the three safety channels required for corrective action (scram) and, in most cases, with concurrence of two out of three of the control channels required for any action by the control system. Flow, temperature, computed thermal power, and measured neutron flux are the parameters utilized; three channels are provided in the control system for each of these parameters. Similarly, three separate channels for each parameter are provided in the safety system. In order to have available reactor power-level information which covers the range from shutdown to full power, three channels using automatically positioned fission chambers are furnished. Each of these instruments provides a signal which is displayed and recorded, without range switching, on a single scale covering a range of ten decades. Each of the instruments also provides period information and serves as a startup channel.

The use of the three-channel coincidence circuits reduces false shutdowns due to instrument malfunction to a minimum and has the very great advantage of permitting each channel to be tested in turn without disturbing the operation of the reactor.

Control of reactivity is exercised by the control system through the action of electric motors on the five control plate drives. As described in Sec. 5.2.4, the control elements are in the form of two concentric cylinders which are located between the fuel and reflector regions of the core. The control drives enter the reactor vessel through seals from the subpile room beneath the vessel. The inner control plate, which is a complete cylinder, is moved upward into the reactor to decrease reactivity; its motion is accomplished only by motor drive. The outer control cylinder, which is divided into four quadrants, is moved downward into the reactor to decrease reactivity. Each quadrant is equipped with a reversible electric motor and a fast insert air motor attached to the same drive and a fast-acting scram latch.

The inner control cylinder, which has no fast shutdown feature, is motor driven to provide both shim and regulating action. Separate motors and gear boxes are provided for the shim function and for the regulating function. The outer control plates, called the shim-safety plates, are used for shim control. Release of the magnets which hold the scram latches in position results in the rapid insertion of the plates under the action of gravity, water flow, and an acceleration spring. The shim-safety-plate drives and scram latches are modified versions of those installed in the ORR in 1964.

In referring to the motion of the control cylinders, later in this section, the following nomenclature will be used. The word "insert" will always mean motion of a control plate in the direction which decreases reactivity. This is upward in the case of the inner whole cylinder and downward in the case of the quadrants. The word "withdraw" will always mean motion of a control plate in the direction which increases reactivity, downward in the case of the inner plate and upward in the case of the outer plates. The outer plates are designated shim-safety plates 1 to 4. The inner plate is designated the shim-regulating plate; however, because it has two functions, it will be discussed as if two, rather than one, control elements are installed and will be called the No. 5 shim plate, or the regulating rod, depending upon the function under consideration.

Normally, the five movable control elements are operated in concert in order to maintain symmetry of the poison distribution in the vertical and azimuthal directions. Provision is made for individual plate movement where needed for trimming the plate positions and for certain special operations and tests.

Several degrees of corrective action resulting in control plate motion are available to the control system in order to permit it to cope with abnormal situations. The safety system action results in a fast scram by interrupting power to the safety plate magnets.

As previously indicated, certain process system parameters are utilized as integral parts of the reactor safety and control systems. The process instruments which fall into this category measure the primary coolant flow, primary coolant inlet and differential temperatures, the primary coolant inlet pressure, and the reactor thermal power. The instruments for each function are furnished in triplicate in order to be consistent with the overall two-out-of-three coincidence system and the necessary isolation of safety and control systems. Thus, there are six channels provided for each parameter, three for the control and three for the safety system. Other process instrumentation which is not directly concerned with the control and safety systems utilizes conventional components of modern design and is provided singly or in duplicate as necessary to ensure continuity of operation.

The control area consists of two rooms, the control room and the amplifier and relay room (called the auxiliary control room on many construction drawings). The control room (Fig. 8.1.2) is located above the amplifier and relay room, and a stairway for personnel access connects the two. Access for interconnecting wiring is provided through conduits and floor slots.

Cabinet *A*, at the extreme left, is provided for future use by the critical facility controls. Cabinets *B* to *D* each house portions of one startup channel, one safety channel, and one control channel. Cabinet *E* contains the main process system graphic panel. Cabinets *F* to *I* are graphic panels serving the liquid waste system, gaseous waste system, health physics monitoring system, and electrical system respectively.

The console is the control point for startup and overall system operation and surveillance. Operation of essentially all the process system is handled at the graphic panel in the control room. The safety system tests, to be described later, can be made by the operator from the console while observing the results on indicators on cabinets *B* to *D*.

Instrumentation not requiring direct observation or manipulation by the operator is located in the amplifier and relay room. This includes such equipment as control relays, signal transmitters and modifiers, high-power servo amplifiers, etc. An area in this room is set aside for spare equipment and for an on-line computer.

Communication between the control room and other areas of the building is by means of an intercom system and a public-address system. The master control for the intercom system is located in the control room and can be operated from the console. If operational problems require the undivided attention of the operator, he may transfer the master communications control station duties to the operation supervisor's room, located adjacent to the control room. These systems are supplied with power from the emergency-power supply.

The most vital information concerning the operation of the system is presented on cabinets *B* to *E*. Accordingly, these cabinets are in the most favorable location to command the attention of the operator. Similar information on the three channels of safety, startup, and regulation are presented at similar positions on cabinets *B* to *D*. The graphic panel, cabinet *E*, is incorporated in the process system to provide for the rapid assessment of the status of the plant process system and to allow adjustments to be made as necessary.

Information from the four radiation monitoring systems is also available in the control room in the form of annunciators, indicators, and recorders. The four systems are

1. a health physics monitoring system, consisting of a number of monitors located in various parts of the building which measure the beta-gamma radiation level and the particulate activity level;
2. a gaseous-waste monitoring system, which measures the activity of the off-gas and airflow passing through the exhaust stack;
3. a liquid-waste monitoring system which measures the activity level of the several liquid-waste systems;
4. a coolant activity monitoring system, which is intended to provide information regarding the activity level in the several coolant systems.

In addition to providing information to the HFIR operations staff, the gaseous- and liquid-waste monitoring systems also transmit information to the ORNL central waste monitoring system, which is located in the main part of the Laboratory (Building 3105) and which provides overall monitoring and handling of all gaseous and liquid waste in the ORNL area.

Information from the health physics monitoring system is also transmitted to the ORNL Emergency Control Center, which coordinates emergency measures for the Laboratory in the event of a serious incident.

The annunciators for all systems are located above the relevant control room cabinets, as shown in Fig. 8.1.2.

The power level, as indicated by one of the wide-range counting rate instruments, is indicated at various points throughout the building to provide information on the operating level of the reactor.

8.2 Control Plate Drive Mechanisms*

8.2.1 Introduction

Because the control plate arrangement in the HFIR is somewhat different from that usually encountered, and because a knowledge of how the control plates function is essential in order to understand the workings of the control and safety systems, the control drive mechanisms will be described before embarking on a discussion of the control and safety systems.

As previously described, control of the reactor is effected by the movement of two concentric cylinders located between the fuel and reflector regions of the reactor (Fig. 5.1.1). This configuration was chosen in order to minimize the perturbation of the power distribution caused by the withdrawal of the control plates as burnup of the core occurs. The control elements are moved by mechanisms located in the subpile room beneath the reactor and shielded from it by water, steel, and high-density concrete (see Sec. 9.2.3). The connecting elements between the drive mechanisms in the subpile room and the control elements in the reactor vessel are drive tubes of stainless steel which extend upward from the subpile room through seals in the reactor vessel lower head.

The inner cylinder, the shim-regulating plate, moves downward to increase reactivity. Because this is the direction of gravitational, hydraulic, and flow forces, this control element is bolted securely to the drive tube and moves only when motor driven; it has no provision for fast scram action. When used for shimming, it will be referred to as the No. 5 shim plate, and when used for regulation, as the regulating rod. The outer control cylinder is divided into four quadrants, each of which is provided with a separate control drive mechanism and a quick-acting scram latch. These plates, shim-safety plates 1 to 4, move downward in the direction of the gravitational, hydraulic, and flow forces to decrease reactivity. A view which shows the overall arrangement of the control plates and drive mechanisms is given in Fig. 8.2.1(*a*).

8.2.2 Shim-Safety Drives

The safety release mechanisms are of the same general type as those used in the ORR and the ETR (i.e., the ball-latch type), as shown in Fig. 8.2.1(*b*). This mechanism consists of a latch head located near the top of the drive tube, in the reactor vessel, with a latch push rod extending downward into the subpile room through the center of the drive tube; a seal between the push rod and the drive tube is provided at the lower end of the drive tube. This latch push rod 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 currents in the magnet 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 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 push rod. When the balls are extended, the latch is "cocked" and is held in position by the electromagnet. A hydraulic cylinder assembly allows for thermal expansion of the push rod, but forms a solid connection when the push rod makes the rapid movement to initiate a scram.

Concentric with the drive tube, but entirely within the reactor vessel, is a hollow control-plate extension tube to the top of which the bottom of the shim-safety quadrant is attached by means of a self-aligning coupling. Inside of this control plate extension tube, 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 extension tube, an upward movement of the drive tube will lift the extension tube and thus

*J. E. Jones, Jr., *Control Element Drives for the High Flux Isotope Reactor*, ORNL-CF-67-11-34 (Nov. 10, 1967).

raise or withdraw the shim-safety quadrant. If, while the shim-safety quadrant is raised, the magnet is de-energized, the latch push rod is moved downward by pressure forces and springs, the balls retract, and the control plate and its extension tube are free to move downward, reducing the reactivity.

The lower end of the control plate extension tube forms a shock-absorber piston which enters a shock-absorber cylinder near the lower end of the stroke to decelerate the safety quadrant and extension tube following a scram.

The design criteria specify that the shim-safety plates start moving within 0.01 sec after a safety parameter is exceeded. In addition, the travel of the safety plates following release shall be equal to, or greater than, that which would be obtained if they had an initial acceleration of $4 \times g$ which decreased to $1 \times g$ at the end of 6 in. of travel.

The scram latch mechanism is designed to be the coincidence element in the two-out-of-three scram circuit. The output of each safety channel is the controlled current in one of three electrically independent coils in each scram-latch magnet. These magnets are therefore the elements which determine the behavior of the scram latches on the basis of the outputs of the three safety channels. A magnet of this type is shown schematically in Fig. 8.2.2, together with curves of the magnetic fluxes in the various portions of the circuit as a function of the excitation ampere turns. The first increment of excitation is provided by any one safety channel and produces flux only in the yoke and shunt, the air-gap reluctance being high enough so that the armature flux is negligible. As the excitation is increased, corresponding to the output of two safety channels, the magnetic material in the shunt reaches saturation, and the additional excitation increases the flux in the armature. Finally, at the still higher excitation produced by three safety channels, the yoke, armature, or both saturate. The resulting force on the armature is quite small when the shunt is unsaturated and rises rapidly (approximately as the square of the excitation) from nearly zero at the shunt-saturation point to full force at the armature-saturation point. By the proper choice of coil turns and current, it has therefore been possible to develop a magnet which exerts little holding force if one coil is energized and exerts the full force if two or three coils are energized. This magnet is thus a two-out-of-three coincidence element and will release if any two of the safety channels are de-energized but will hold if only one is de-energized. As will be discussed in Sec. 8.4.2, one of the advantages of this system lies in its adaptability to on-stream testing.

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 rotation of the nut. The nut for each drive is driven through 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 5.75 in./min with the drive motors operating steadily at their maximum unloaded speed.

The scram plate drives are also provided with a nonreversible air-powered motor coupled through appropriate gearing to provide for unidirectional high-speed insertion of the drive following a scram or when needed during an ac power outage. The design speed is approximately ten times the speed developed by the electric motors, or about 60 in./min [see Sec. 8.3.2(*a*)].

All drives are provided with position-sensing and -transmitting equipment which is used to transmit information to the control room for use in the control system and for the operator's information. In addition, a variety of limit switches are provided as necessary to indicate and limit drive position, latch operator position, etc. Seat switches are also provided to indicate independently of the drive mechanisms when the shim-safety plate is fully inserted.

8.2.3 Shim-Regulating Cylinder Drives

The shim-regulating drive for the inner control cylinder does not have a safety release mechanism. The drive system for this control element consists of a drive tube which extends from the subpile room, through a seal in the vessel lower head, to a support ring, which is the lower part of the inner control cylinder. This support ring is fastened to the upper end of the drive tube by a coupling which allows for some misalignment between the control cylinder and the drive tube.

The drive gearing provided for shim motion of this unit is similar to that for the four outer plates; however, no safety release mechanism or fast run-down air motor is provided. The design speed is the same as for the outer rods (i.e., 5.75 in./min). The system is provided with limit switches, position sensors, and transmitters similar to those for the outer drives.

The inner cylinder has a regulating drive of limited stroke which operates independently of the shim drive to move the shim-regulating cylinder under automatic control. As described in Sec. 8.5.3, the regulating system is unusual in that three independent channels are provided, the output of each being the angular velocity of a motor shaft. The three channels are combined in a servo input gear box which has a single output shaft driving the nut of a ball-bearing lead screw which has a limited stroke. The servo input gear box contains two differential gear mechanisms which provide the velocity addition required to cause the shim-regulating plate to be moved at a velocity proportional to the algebraic sum of the velocities of each of the three servo motors. The input gearing for each motor is of the worm–worm wheel type and is self-locking so that a de-energized motor will not be rotated by the other two motors or by the load. The servo motors are of the low-voltage printed-circuit type and for normal rated output require 48 v dc at 16 amp.

Design speed of the regulating motion with all three motors driving in the same direction is 15 in./min with the stroke limited by mechanical stops in addition to the limit switches provided for normal operation. The regulating drive position is transmitted to the control room in addition to the shim drive position in order to enable the operator to supervise the servo systems. The shim drive position transmitter always provides information on the plate position; however, the regulating drive position indicator only supplies the position of the drive mechanism.

In addition to the drive gearing and motors for both shim and regulating functions, the central plate drive is provided with a "backup cylinder." This oil-filled hydraulic cylinder is attached to the lower end of the shim drive lead screw and has the primary function of limiting the rate of travel of the central drive rod in the unlikely event of complete failure of the rotating nut or regulating drive gearing. As a secondary function, the hydraulic cylinder is used to balance out most of the force caused by the coolant pressure in the reactor vessel being transmitted through the drive tube to the mechanisms. This force balancing serves to reduce the loading on the gears, thereby reducing the wear and also allowing better operation of the automatic power regulation system. The force balancing is accomplished by having the hydraulic cylinder piston rod approximately equal in area to the guide tube emerging from the reactor vessel and by operating the hydraulic cylinder at the same pressure as that in the reactor vessel. This pressure is obtained by interconnecting the cylinder and the reactor vessel through a double bladder-type accumulator system, which requires no instrumentation for operation and which will maintain the same pressure even if the accumulator bladders should leak. Two accumulators are provided in order to have a buffer zone of water between the reactor water and the hydraulic oil so as to minimize the possibility of contaminating the reactor system with hydraulic oil. The speed-limiting feature of the backup cylinder does not depend on having reactor pressure in the hydraulic cylinder since this is accomplished by selection of the clearance between the cylinder wall and the piston.

The hydraulic piston rod area is somewhat smaller than the guide tube area in order that a net downward force be present in normal operation. The downward force is chosen to be sufficient to keep the gears of the gear drives loaded and thus minimize backlash for good servo operation.

8.3 Control System

8.3.1 Introduction

As previously pointed out, one of the main objectives of the HFIR operation is to continuously maintain the reactor at the highest 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 and compute the heat power 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 rapid so that fast safety action will be only rarely necessary. The required speed of response is obtained by utilizing the accurate, but delayed, heat-power information to continually and automatically calibrate neutron-flux measuring devices by adjusting the gains of the flux amplifiers. The so-called "reset flux" obtained in this way is then employed directly as the basic control parameter.

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. Four degrees of corrective action are available to the control system in order to permit it to cope with abnormal situations with a minimum of disturbance in operation. In order of decreasing severity, these are: (1) *Fast reverse* – air motor insertion of the shim-safety quadrants; (2) *Reverse* – a motor-driven insertion of all control plates; (3) *Setback* – a reduction in the automatic control power demand; and (4) *Withdraw Inhibit* – an action which merely prevents further withdrawal of the control plates. In addition, a slow scram is provided to cause insertion of the shim-safety plates by cutting off power to the magnet amplifiers. Control system action is summarized in Table 8.3.1.

Table 8.3.1. Control System Action

Action	Cause
Fast reverse (air motor)	Flux to flow >1.1 (two out of three)
Reverse	1.1 N_F heat-power signal in *Run* condition (two out of three)
	1.1 N_F flux signal in *Run* condition (two out of three)
	1.5 N_L heat-power signal in *Start* condition (two out of three)
	1.5 N_L flux signal in *Start* condition (two out of three)
	Period less than 5 sec on wide-range counting channel (CRM) in *Start* condition (two out of three)
Setback	Always, when in *Start* condition or key switch *Off*
	Demand-to-flow ratio greater than unity
	Control plate asymmetry greater than 1.2 in. for shim-safety plate or 2.4 for shim-safety plate vs shim-regulating cylinder
Shim withdraw inhibit (mode 1 only)	Key switch off
	Regulating rod not in withdraw limit[a] in *Start* condition
	Log CRM period <30 sec (one out of three) in *Start* condition
	Log CRM not in confidence[b] (two out of three) in *Start* condition
	Shim insert request in either *Run* or *Start*
	Shim in withdraw limit in either *Run* or *Start*
	Following slow scram (two out of three), No. 5 shim plate only
	Following fast scram by safety system (two out of three), No. 5 shim plate only
	Manual inhibit

[a]Affects only automatic startup.

[b]By confidence is meant that the amplifier switch is in the *Operate* rather than the *Test* position, the fission chamber drive controls are in automatic withdraw, and the counting rate is greater than 10 but less than 50,000 counts/sec.

The control system block diagram is shown in Fig. 8.3.1. No attempt is made here to describe the routine features such as limit switches, etc., the utility of which should be obvious; however, certain of the more significant instrumentation is discussed in some detail subsequently.

In addition to the three ionization chambers which provide information for the regulation system, three wide-range counting channels provide flux level and period information for use during startup as well as during power operation. The control function of these channels is limited to the startup condition; they provide for automatic corrective action upon receiving appropriate period and flux level information.

8.3.2 Modes of Operation

Two conditions, *Start* and *Run*, are provided. Changeover from the *Start* to the *Run* condition is manually made when the reactor reaches N_L – approximately 10% of full power. This represents the point at which the heat-power instrumentation starts to yield reasonably accurate information. Moreover, 10% of full power is the level above which the cooling requirements exceed that provided by the shutdown coolant system described in Secs. 6.2.1(*d*) and 6.5.

In addition to the normal sequence required for routine startup and operation (mode 1), two other modes of operation are available in order to provide for versatility and to minimize the necessity of making temporary changes to the system which must later be corrected. These modes (Nos. 2 and 3) permit operation without pressurization but with shutdown coolant flow, and operation with neither pressurization nor coolant flow, respectively.

(a) Operation in Mode 1. – Mode 1 is the normal operating sequence which is used for startup and operation at full power. In this mode, the reactor is pressurized and the coolant system is functioning normally. When in this mode, the startup must proceed automatically since no provisions are made for manual startup.

In the *Start* condition, the shims are withdrawn automatically in concert as long as the regulating rod remains in its withdraw limit and the wide-range counting channels are operating, as indicated by a set of confidence contacts, and are indicating a period of 30 sec or longer.

Although the speed of withdrawal of the five shim plates is such that the average rate of reactivity increase is only about 0.075%/sec in the range from shutdown to criticality, the maximum rate may be as high as 0.25%/sec because of the nonlinear relation between plate position and sensitivity. As explained in Sec. 8.8.3, a startup accident at this rate can easily be handled by the close coupling of the reactivity effects with the fuel temperature. The customary intermittent-withdrawal feature was removed when it lengthened a cold startup by an hour, and it was determined that proper adjustment of the timing cycle would negate any of the questionable protection afforded.

The increase in power during automatic startup is terminated when the reactor reaches approximately 10% of full power (N_L) by servo insertion of the regulating rod. If the servo fails to insert the regulating rod, a reverse is automatically initiated at 1.5 N_L.

When the *Start* phase is completed, the reactor remains under regulating system control at N_L until the operator manually requests a change. With full coolant flow, at 10 Mw the coolant temperature rise from reactor inlet to outlet is $\sim 4°F$, and the heat-power calculators have sufficient resolution to allow accurate comparison of the heat power and flux. After a delay sufficient to allow operator and instrument orientation, the *Run* condition may be obtained, provided that the power level is at N_L, the regulating rod is not in its withdraw limit, and the reactor period is essentially infinite. The operator may then raise the reactor power by increasing the setting of the servo demand. The regulating system will withdraw the regulating rod and thus increase the power of the reactor until it equals the demand. The rate of increase is self-restricted to <4.5 Mw/sec. The demand setting is restricted automatically to a value no greater than that allowed by the available coolant flow, and no increase in demand can occur unless the control plates are symmetrically arranged.

This maximum allowable control plate asymmetry is 1.2 in. for the four shim-safety plates with respect to each other, and 2.4 in. for any shim-safety plate with respect to the shim-regulating cylinder. Greater asymmetry may cause undesirable peaking of the power density and must be avoided when operating at high power. A comparison of the positions of the shim-safety drives is made using voltages generated in potentiometers driven by the plate-drive lead screws. Comparison of the four shim-safety drive positions with that of the shim-regulating cylinder is made in a like manner. If either of these comparisons shows asymmetry greater than the specified amount, a setback in power to N_L is initiated. Because it is the drive positions rather than the actual plate positions which are used, this comparison is invalidated by an indication that one or more of the shim-safety plates is not connected to its drive.

When the regulating rod approaches the end of its stroke, the shim plates must be adjusted. Automatic shim insertion is employed; however, as in other ORNL reactors, shim withdrawal is permitted only under the supervision of the operator. This limitation is at present dictated by safety considerations. The regulating rod worth is limited to a value of reactivity which can be safely inserted as a step function. In order to prevent greater amounts being inserted automatically by the servo, the shim withdrawal must be made only when justified by explainable losses in reactivity. Because of the complex interrelation of variables affecting reactivity in the HFIR core, an on-line computer, which accepts signals from the sensory instrumentation and calculates the required shim action, is being installed to evaluate its use. Meanwhile, manual shim withdrawal, subject to the limitations imposed by the control and safety systems, will continue.

In order to expedite restart following a scram or to provide rapid reactivity reduction during a flow coastdown on electricity failure, fast shim-safety-drive insertion is provided by the use of unidirectional air motors. Considerable care has been taken to ensure that these high-speed motors are incapable of withdrawing the shims. Normally, the fast insertion will be started only when the affected plate is in its seat and the scram latch is disengaged.

One condition, however, does require fast shim insertion to help control the reactor. This occurs during a normal-power outage when ac power is lost to the primary and secondary coolant pumps. In this case, it is desirable to continue to operate the reactor utilizing the emergency coolant flow but with the power reduced to approximately 10 Mw. Fast insertion is necessary to prevent a high flux-to-coolant-flow ratio from tripping the safety circuits as the pumps coast down. This fast insertion is provided by the air motors because, on loss of ac power, the electric shim drive motors are without power for a short time until the diesels accept the load. When the flux-to-flow ratio exceeds 1.1 in two of three channels, the fast-insertion drive will insert the shim plates until the error is reduced.

When steady-state power at N_L has been achieved, the regulating rod will be near its insert limit. This regulating rod position is desirable, because it makes available to the servo nearly the maximum allowable positive reactivity for use in overriding xenon poisoning. The negative reactivity inserted by the shims during power reduction must also be compensated for. These requirements must be handled by the servo until the diesel-driven generators are available to supply power to the shim drive motors, thus enabling the operator to withdraw the shim-safety plates and the shim-regulating plate. The regulating rod will reach its withdraw limit in 30 to 45 sec, provided no shim withdrawal is initiated. If the diesel startup is orderly, ac power to the shim-rod drives will have been restored before this time (see Sec. 10.1.2). The reactor power level can then be maintained at 10 Mw until the available excess reactivity is consumed by xenon growth at a negative reactivity rate estimated to be 0.2%/min. Should the diesel start be delayed, the reactor will go subcritical after the regulating rod reaches its withdraw limit. If the delay is prolonged, the reactor will drop from the *Run* condition to the *Start* condition when the power sags below $\sim$300 kw. The decision whether to attempt a restart must be taken on the basis of the time in cycle when the shutdown occurs and the duration of the shutdown.

(b) Other Modes of Operation. – Two other modes of operation are provided. Mode 2 permits startup and operation without pressurization of the primary coolant system. It is required, and enforced by interlocks, that at least shutdown coolant flow be maintained. The reactor power level is limited by the safety system to a level consistent with the available coolant flow and the low pressure. The control system will remain in the *Start* condition, but automatic group withdrawal of the shims is blocked so that manual operation of the controls is necessary.

Mode 3 permits startup and operation of the reactor without pressurization and, if desired, without any forced coolant flow. The power level will be limited by the safety system (see Sec. 8.4) to a level at which convective flow in and around the fuel is adequate to remove heat – approximately 100 kw. As in mode 2, the reactor will remain in the *Start* condition, and manual control will be utilized.

These two modes of operation were provided in order to facilitate the various tests such as power distribution, control plate effectiveness, and flux distribution investigation, which must be done at low power. Mode 3 is also that in which the initial critical tests were performed. By providing these modes, the necessity for temporary bypasses and changes in the control and safety systems is largely eliminated.

8.4 Safety System

8.4.1 Introduction

The safety system is 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.

Because the safety system is provided to protect the reactor against damage resulting from the power level exceeding the heat transfer and temperature limitations of the fuel, safety action based upon the power exceeding a preset fixed power level is not, in itself, adequate. The safety action must also be related to coolant flow; and for this reason the neutron-flux level at which the safety system will scram the reactor has been made a function of coolant flow. Because heat-power instrumentation, similar to that used in the control system, provides a more accurate measure of the power level than the neutron detection instrumentation, heat-power information from this instrumentation is also used as a scram parameter. Other safety parameters utilized are the reactor inlet temperature, the rate of rise of neutron flux, the main coolant flow, the primary coolant system pressure and the primary coolant system gross activity.

The accuracy required of the HFIR safety system is considerably greater than that required of other ORNL reactors because of the diminished margin between the operating power level and the calculated hot-spot incipient boiling power level. Accordingly, as in the control system, the slow, but accurate, heat-power calculators are used to continually and automatically reset the gain of the neutron-flux safety amplifiers. Independent heat-power calculators are used for each of the three neutron-flux safety channels so that a single heat-power calculator failure can disable only one of the three channels.

8.4.2 Operation of the Safety System

Three independent channels of nuclear and process instrumentation comprise the safety system of the HFIR. Each channel includes a transducer for each of the safety variables – flux, flow, temperature, pressure, and primary coolant activity – and the necessary amplifiers and computers. Each of these channels has associated with it one of the three separate battery systems which provide failure-free power for the vital safety functions (see Sec. 10.1.4). A block diagram of the safety system is shown in Fig. 8.4.1, and the following descriptive material presumes reference to this figure.

As described in Sec. 8.2.2, the output from each safety channel controls the current in one of the three electrically independent coils in each scram latch magnet, thus forming a two-out-of-three coincidence system. Scram trip signals from two of the three channels, even though initiated by different parameters, will cause a scram.

One of the main advantages of utilizing the magnet as the coincidence element lies in the ability to test any one channel during operation and to verify that the appropriate coil in each magnet is de-energized without the test causing a reactor shutdown. Moreover, the test can be applied in turn to each one of the parameters utilized by the channel. During operation, the test is initiated by locally perturbing the parameter so that the transducer is actually exposed to the condition which would be potentially dangerous if it truly prevailed in the process. Thus the entire system (except for the actual dropping of the shim-safety plates) can be tested during operation, channel by channel, from sensor to magnet without inhibiting the ability to scram and without interrupting operation. Moreover, a single inoperative safety channel can be repaired without interrupting reactor operation or compromising safety. A list of the safety parameters, together with the testing method utilized, is given in Table 8.4.1.

Table 8.4.1. Scram Parameters

Parameter	Scram Value	Test Method
1. Ratio of reset neutron flux to flow	>1.3	1*a*. Neutron flux – connect extra ion chamber section 1*b*. Flow – bypass differential pressure cell by opening valve
2. Heat-power	>120 Mw	2. Core temperature rise – spray hot water onto outlet temperature sensor
3. Reactor inlet temperature	$>135^{\circ}$F	3. Spray hot water onto inlet temperature sensor
4. Rate of rise of neutron flux	>20 Mw/sec	4. Same as item 1*a*.
5. Main coolant flow	$<$Shutdown flow	5. Same as item 1*b*.
6. Reactor primary pressure	<375 psig	6. Open blowdown valve at sensor; impedance of tube between sensor and tank gives lower sensor pressure
7. Faulty fuel element detector	$>$Normal (approx. 140%)	7. Rotating shutter

To provide for testing while the reactor is in the shutdown condition, electrical test circuits are also built in. By pushing the appropriate button, the operator can superimpose a scram signal on the transducer and test all the instrumentation except the transducer itself.

The electronic instrumentation for one of the three identical safety channels is shown diagrammatically in Fig. 8.4.2. The instrumentation from which the signals are derived is described in subsequent sections. After suitable amplification, each signal is applied to a Fast Trip Comparator (Q-2609).* This is a bistable device whose aspect is determined by whether the signal represents a *Safe* or an *Unsafe* condition, as established by the setting of the trip level. This Comparator can be made to function in any one of a variety of ways – upscale trip, downscale trip, positive or negative signal, and algebraic difference of two signals – depending upon the interconnections between certain terminals. These interconnections are permanently wired on the drawers into which the comparators are plugged; thus the comparators themselves are interchangeable. The

*Numbers with the prefix Q- are used by the ORNL Instrumentation and Control Division to identify various specific components and provide an unambiguous reference to the component.

comparators supply normal output signals of -10 v, which are changed to 0 v within 200 μsec after the input signal passes into an *Unsafe* region. Readout relays and pilot lights are also provided.

The output signals from the seven comparators of each safety channel are utilized as input to the OR gate (Q-2612) for the channel. The output potential of the OR gate is approximately -10 v if all its input signals are -10 v, but is 0 v if any input is 0 v.

The output signal from the OR gate turns on and off the regulated currents to the magnet coils; for each coil the necessary amplification and regulation are provided in a Magnet Control Amplifier (Q-2613). The magnet current changes less than 1% for a 28- to 36-v variation in (battery) supply voltage or for ambient temperature changes between 20 and 60°C.

The magnet current is turned on and off by means of a transistor which is normally saturated by the normal -10 v input signal supplied by the OR gate. When the input signal voltage goes to zero as a result of a scram request, the output current of the Magnet Control Amplifier goes to zero. For a resistive load, the load current can be turned off by this amplifier in less than 50 μsec; however, the inductive emf of the magnet when the amplifier is turned off will cause a current to flow through the Zener diodes, which limit the surge voltage to protect the output transistor. Experiments with various inductive loads, including the developmental rod-release magnet, have shown that the current in the magnet coil is reduced to zero within about 3 msec of a step change in input signal voltage from -10 v to zero. Persistent eddy currents in the yoke and armature of the magnet continue to supply sufficient flux to hold the armature for another 2 msec or so, and the overall time to first movement of the armature is about 5 msec.

The safety system borrows many techniques from analog computers. It can be seen from Fig. 8.4.2 that a basic building block is the Operational Amplifier (Q-2605). This is a commercial unit purchased to meet ORNL performance and packaging specifications. It is chopper-stabilized with an open-loop gain of $>10^7$ and uses transistors throughout. Several years of experience with a large number of these elements, in computers and on-line process-data handling systems, gives confidence that their performance and reliability are appropriate for reactor control and safety use.

8.5 Nuclear Instrumentation

8.5.1 Introduction

The important instrumentation systems employ redundant isolated channels so that no single component failure will render the reactor inoperative. The three channels each of safety, startup, and regulating instruments receive their power from three independent banks of station batteries, each bank serving a single channel in each system. Each bank consists of a sufficient number of nickel-cadmium cells to produce +32 and -32 v. An individual charger capable of carrying the full load current of the connected instruments plus the additional current to recharge the batteries from the fully discharged condition within 8 hr is provided for each bank. The batteries are unfused since nickel-cadmium cells are not damaged by short circuits.

All the nuclear instrumentation equipment utilizes solid-state devices (transistors, diodes, etc.) rather than vacuum tubes in order to take advantage of their potential for increased reliability. The more important of the nuclear components of the control, safety, and startup instrumentation will be discussed in this section. It will be recalled, however, that the HFIR is controlled to a large extent by process variables such as flow, temperature, and pressure. The instrumentation utilized to accomplish this will be discussed, together with other process instrumentation, in Sec. 8.6.

8.5.2 Safety System Instrumentation

The major nuclear components of the HFIR safety system are the ionization chambers, which detect and transmit a signal proportional to the neutron flux, and the flux amplifiers, which amplify and feed the signal to the trip comparators. The action of the trip comparators and OR gates which activate the scram latches has already been discussed in Sec. 8.4.2.

(a) Ionization Chambers. – The HFIR utilizes three multiple-section ionization chambers (Q-2618), each of which serves a single safety channel and a single regulating channel. Three of the four sections of each chamber are designed to provide neutron-flux signals: one to the servo, one to the safety system, and one for the safety system test. The fourth section is provided to obtain a signal proportional to the gamma flux. An exploded view of the chamber is shown in Fig. 8.5.1. The active sections, contained in the inner housing near the front of the chamber, are electrically independent or interconnected, as required (Table 8.5.1). Although the gas filling is shared by all four sections, the safety and control functions are strictly independent electrically.

Table 8.5.1. Ion Chamber Sections[a]

Position	Function	Interconnection	Neutron Sensitivity[b] (amp/nv)	Gamma Sensitivity[b] (amp-hr/r)
Front	Regulating (servo)	Common high-voltage connection	4.0×10^{-15}	1.2×10^{-11}
	Gamma		0	2.5×10^{-12}
	Safety	Common high-voltage connection	2.6×10^{-15}	1.4×10^{-12}
Rear	Safety test		1.2×10^{-15}	0.8×10^{-12}

[a]Filling gas: nitrogen at 1000 torr.

[b]Referenced to thermal neutron and gamma flux at center of gravity of the servo section (i.e., corrected for attenuation in the chamber).

The gamma section of the chamber is not at present connected to the reactor control system. It is included in order to obtain information which may later permit the use of gamma as well as neutron flux for the control or safety systems.

The ion chambers are located in thimbles which penetrate the reactor vessel. The incentive to develop these multisection chambers arose largely because of the high cost of such thimbles and associated chamber-positioning devices and the scarcity of locations which are both accessible and relatively unperturbed by changes in beam-hole experiment configurations.

Connections to the active sections are made by special ceramic-insulated conductors which pass through an aluminum scattering and shielding block to a connector in a welded bell housing. The connector is sufficiently radiation resistant with the shielding provided. From the bell housing, shielded and radiation-resistant flexible cables pass through a welded, waterproof, flexible, stainless steel shell to a junction box located in a dry position behind the pool parapet.

The safety chamber output for normal full-power operation is usually in the range of 20 to 50 μamp, the corresponding scram point being 1.3 times greater. The saturation characteristic of the chamber is such that it is still 97% saturated at a 500-μamp output current with the normal chamber high voltage of 250 v.

The chamber assemblies are held in the reactor vessel thimbles with the remotely adjustable positioners shown in Fig. 8.5.2. The drive mechanisms position the ion chambers as required over a range of 12 in. The drive mechanism assembly is supported from the pool wall by a vertical mast which is also used to bring the electrical leads above the surface of the pool for routing to the junction box. A flexible shaft permits manual operation of the drive mechanism from above the water level.

(b) Neutron-Flux Amplifier. – Each of the safety channels is equipped with a flux amplifier (Q-2602), which receives a signal proportional to the neutron flux from the ionization chambers and amplifies this current so that it can be utilized in the safety system.

The availability of the field effect transistor has made possible the development of this low input leakage-current operational amplifier. The back-biased gate electrode of the input transistor has a dc leakage input current of less than 5×10^{-11} amp; and, by proper choice of the feedback resistor, the amplifier output voltage corresponding to the trip level of approximately 10 v can be related to any value of current from 10^{-8} to 10^{-4} amp. In the HFIR, a value of 1.5×10^5 ohms for the feedback resistor would correspond to about 50 μa at 100 Mw (nominal maximum ion chamber output at full power) if the feedback resistor were connected directly to the amplifier output; in fact, there is an intervening voltage divider to provide variable gain. In addition, provision is made for remotely changing the feedback resistor in order to change the sensitivity of the channel as required for operation in modes 2 and 3.

Temperature compensation for the input transistor is provided so that the measured overall temperature stability is 0.3 mv/°C at the output, with the input open-circuited, for any feedback resistance between 10^5 and 10^9 ohms. The rise time for a step increase in input current is <300 μsec for normal scram level with approximately 2×10^5 ohms feedback resistance and is ~5 msec with a 2×10^8 ohms feedback resistor, which is the value used for mode 3 operation. The output of the amplifier will drive to ±12 v a load of 300 ohms in parallel with 0.05 μf.

The Flux Amplifier Module (Q-2602) also contains a high-voltage supply for polarizing the ionization chamber. The supply consists of a 25-v preregulator followed by a dc-to-dc converter.

(c) Flux Reset Feature. – At full power the neutron flux at any ion chamber location may vary over the lifetime of a core. Differences between individual cores may account for an additional slight variation. Because of the small margin between the 100-Mw operating power and the 130-Mw neutron-flux scram, the gains of the flux amplifiers would require adjustment during a single core lifetime. The method frequently used to make this adjustment, by either manually changing the amplifier gain or moving the ion chambers on the basis of a single heat-balance measurement, is considered inadequate for the HFIR, because it is considered undesirable to permit a single computational error to result in mis-setting of all safety channels. This is especially true because these changes in gain require frequent and routine adjustments. Accordingly, the output of each heat-power calculator is used to adjust the gain of a flux amplifier so that the reset flux signal will correspond to the actual heat power. The heat-power measuring equipment and the gain-resetting mechanisms are required to have the same order of redundance, independence, testing, and reliability as the neutron-flux safety instruments themselves. It is worth noting again that all safety channels on the HFIR fulfill these requirements.

The resetting of the flux amplifier gain (Fig. 8.4.2) is accomplished as follows: The gain of the flux amplifier is controlled by a variable voltage divider in the feedback network. The output of the flux amplifier is compared with the heat power calculated from flow and inlet temperatures; the amplified difference drives a servomotor, which positions the variable voltage divider in the feedback network of the flux amplifier to reduce this difference. The maximum motor speed is slow – ~5 min to cause the full gain change of a factor of 1.33 – so that no significant change can take place during a maneuver and especially during an incident. Independent heat-power calculators and reset mechanisms for each set of instruments meet the criteria of redundancy and independence.

The scram at 120-Mw heat power has the accuracy of the heat-power instrumentation; the scram at 130-Mw reset neutron flux at full flow has the fast response (trip in <1 msec) of the nuclear instrumentation plus the accuracy of the heat-power instruments as gained by use of the flux reset; and the two complement each other. Both are necessary although they are redundant in terms of accuracy.

(d) Afterheat Correction. – Immediately after a substantial reduction in reactor power, fission product beta- and gamma-heating can produce a significant fraction of the new, lower reactor power. As an example, the controlled reduction in reactor power from 100 to 10 Mw can be completed in less than 1 min, at which time the decay heat power is ~5 Mw. Assuming constant effective ion chamber sensitivity during the maneuver, the reset flux signal might be 10 Mw, apparently safe. However, the true power generation in the core under these conditions is 15 Mw, an undesirably high value.

In order to cope with the afterheat problem, the complex behavior of fission product beta- and gamma-heating is approximated with a signal with two first-order time constants (20 and 200 sec) which is added to the flux signal. This approximation is adequate because the flux reset from heat power will make longer-term corrections since the reset flux is made to equal the measured heat power, which includes the gamma heating. The regulating system must take afterheat into consideration in the same way (see Sec. 8.5.3).

(e) Flow and Rate Scrams. – As a result of extensive studies of the thermal and nuclear behavior of the core (see Sec. 8.8), it has been determined that for a large class of accidents (all except the very fastest at full power), no core damage will result if the neutron-flux scram point is set at $1.3\ \Phi_r/F$, where Φ_r is the flux reset corresponding to the actual heat power at equilibrium, and F is the primary coolant flow (never less than shutdown flow). The quantities Φ_r and F are expressed in percentages of their values at full power and full flow, so that at 100 Mw with a normal flow, $\Phi_r/F = 1$. As shown in Fig. 8.4.2, this is accomplished by using a fast-trip comparator in the differential mode so as to trip when $\Phi_r > 1.3\ F$. The offset of the comparator is less than 0.5%.

The function Φ_r/F is generated in a quarter-square analog multiplier and is displayed in the control room in order to inform the operator of this ratio while the reactor is running.

It is customary to provide a period scram, and the work of Tallackson *et al.*[1] and Ditto[2] has shown that for most reactors a substantial extension of safety system performance can be gained in this way under many circumstances. The HFIR is atypical in this respect because of its very strong self-shutdown feedback and small margin between design power level and incipient boiling power level. Power excursions initiated at high power in the HFIR have potentially much more severe consequences than those initiated at low power. Consequently, a simple $d\Phi_r/dt$ trip will provide the desired extension of safety system range without introducing the disadvantages of presently available period circuits. The trip level is set at 20 Mw/sec, arbitrarily above the maximum 4.5 Mw/sec approach to full power. For operation in modes 2 and 3, the rate trip is reduced in terms of megawatts per second by the same ratio that the flux level trip is reduced. The "derivative" is generated in an RC circuit rather than by computing a true derivative; the main advantage is that noise is less likely to produce a trip.

(f) Testing and Monitoring of the Flux Channels. – An ideal test for the high-neutron-flux scram would be the temporary removal of a neutron absorber from in front of the ionization chamber, thus locally "perturbing the process." This was not considered practical in the HFIR; so as an alternative the ionization chamber is built in two portions, with a common high-voltage connection but separate signal sections. The test is performed by closing a relay contact in the flux amplifier which connects the "testing signal" section of the ion chamber to the normally used signal section of the chamber; the current is thereby increased 40 to 50%, depending on the neutron-flux gradient. By this means, except for ion-chamber saturation, all components of the safety channel, including the signal connections, are tested; the cables to the signal electrodes of the test section and safety section are run in such a way as to check the continuity of the three signal cables as well as the presence of the polarizing voltage.

The "in-process" test described in the previous paragraph, although inclusive and highly reliable, is limited to operation at a significant power level and is not adjustable. For this reason, a group of electrical tests is also provided. A relay-actuated contact connects a source of current to the chamber signal electrode via a separate cable. A successful trip by this test current, which is fixed in value and large enough to initiate a trip under all variations of amplifier gain and reactor power, demonstrates, to a degree, the operability of the entire system. The most obvious use for this test is at zero reactor power, before starting or restarting the reactor. The test current is applied slowly to avoid actuating the rate trip.

---

[1] J. R. Tallackson *et al.*, *Trans. Am. Nucl. Soc.* 3, 428 (1960).

[2] S. J. Ditto, *Trans. Am. Nucl. Soc.* 3, 456 (1960).

Similarly, another relay-actuated contact furnishes a current whose rate of increase is, under all conditions, fast enough to actuate the rate trip.

A continuously operating monitor circuit alarms whenever the chamber polarizing voltage falls below ~200 v.

The above tests and monitors are intended primarily for the use and information of the reactor operator. The instrument technician has at his disposal yet another test circuit, by means of which a controllable current may be added to the chamber signal electrode. This can be used for calibration and trouble-shooting; these controls are on the instrument module, whereas the relay-actuated tests can be conducted from the reactor operating console.

8.5.3 Regulating System Instrumentation

The HFIR is equipped with a multiple-channel power regulation (servo) system which is arranged to minimize the consequences to operating continuity of a single component failure. Three independent controllers are provided (Fig. 8.5.3). The output of each controller is the velocity of a servomotor shaft made proportional, by tachometer feedback, to an appropriately chosen power signal error. The three shaft velocities are added algebraically by a mechanical double-differential mechanism.. The output of this differential in turn drives the regulating rod. Each of the servomotors receives power from one of the three station batteries. The servomotors are of the permanent-magnet printed-circuit type and require 48 v at 16 amp for normal rated output.

Theoretical and experimental studies of the regulating system[3] have revealed that the stability and margin of such a system are identical to those of a single servo, with three times the gain of any one of the three servos; that if one servo jams in such a way that its output shaft cannot turn, then the remaining two will control the reactor with two-thirds the gain of the complete system; and that if one servo fails by running away, the other two will drive in the opposite direction at half the runaway speed, thus retaining the reactor under servo control.

It follows from the foregoing that any failure of a single servo will not jeopardize operating continuity. Moreover, the system is easily monitored by observing the three velocities; failures, particularly of the runaway type, will be quite evident.

The control variable is reset neutron flux as sensed by the servo section of the ionization chambers and reset by the heat-power calculators (Fig. 8.5.4). The system is virtually identical to that used by the safety system, including the afterheat correction, but utilizes three separate instrumentation channels and is completely independent of the safety system. The reactivity available to the regulating system is limited by positive mechanical stops to an amount which can be easily handled by the safety system even if inserted as a step.

The power demand is controlled by the operator. All three power demands are normally moved in concert, but they may be trimmed individually. The demands are clamped at a value consistent with the available primary coolant flow, both to cause an automatic power reduction during flow reductions and to prevent an inadvertent request for a power that is too high for the existing flow. The demands are not automatically increased following a flow increase; this must be done manually.

8.5.4 Wide-Range Counting (Startup) Channels

The entire operating range of the reactor is monitored by three counting channels which use automatic positioning of the detectors to extend the limited range of the more usual arrangement.[4] The general arrangement of one such channel is shown in Fig. 8.5.5. The technique makes use of the variation of neutron flux as a function of detector position which is nearly exponential in most shielding configurations. A function generator is used to make the detector-position signal more

[3] C. H. Weaver (internal memorandum), *Multiple Controllers for a Single Process* (January 1961).

[4] R. E. Wintenberg and J. L. Anderson, *Trans. Am. Nucl. Soc.* 3, 454 (1960).

nearly proportional to the logarithm of the neutron attenuation. Adding this signal to a signal proportional to the logarithm of the counting rate gives a resultant signal proportional to the counting rate multiplied by the neutron attenuation, which is in turn proportional to the logarithm of the reactor power. Computation of a suitable derivative will then yield the reactor period. The chamber is moved by a drive mechanism under the control of a small servo system whose function is to attempt to maintain a constant counting rate near 10^4 counts/sec.

When the reactor is shut down, the counting rate is much less than 10^4 counts/sec; and the chamber servo will drive the detector to its innermost position. As the reactor is started up, the counting rate increases slowly, causing the indicated reactor power to increase correspondingly. As the reactor proceeds to higher power, the counting rate reaches 10^4 counts/sec and the servo withdraws the detector, thus keeping the counting rate approximately constant; the change in detector position is now also used to indicate the increased reactor power. The servo need only be fast enough to follow normal maneuvers; any transients which are too fast for the servo will change the counting rate so that the channel will read correctly in spite of the lagging servo.

The advantages of this technique are the absence of any necessity for range switching or any other action by the operator, the wide range covered, the elimination of the "dead time" induced when the chamber is withdrawn, and the constant optimum counting rate of the amplifier when sufficient flux is available.

The detector and preamplifier, together with appropriate radiation-resistant interconnecting cable, are contained in a flexible assembly.[5] This assembly has a maximum outer diameter of $\frac{3}{4}$ in. and is made flexible so that a guide tube or thimble, curved as required, can be installed in the reactor shielding leading to the core. The assembly is waterproof and can operate continuously in an ambient temperature of 100°C, surrounded by either air or water.

In the HFIR installation the thimble is located below the reactor core and runs vertically to the bottom flange of the reactor pressure vessel. This permits the fission chamber assembly to be installed and removed from the subpile room. The fission chamber can be inserted to within $21\frac{1}{2}$ in. of the horizontal center line of the core. Below this point 6 ft 8 in. of travel is provided by a rack-and-pinion drive unit. The unit is driven by a dc motor, providing a variable travel speed in either direction with a design maximum speed of 60 in./min. The overall travel of the unit is restricted by insert and withdraw limit switches. The withdraw limit switch is actuated just before full power is reached, to reduce unnecessary wear of the mechanical components of the servos during long, continuous full-power operation. In case these fail, the stroke is further limited by positive mechanical stops.

The detector is a fission chamber with concentric cylindrical electrodes coated with 1 mg/cm^2 of U^{235} over a total area of 30 cm^2. The chamber is well saturated with 150 v applied; the operating polarizing potential is 270 v. The collection time is 80 nanosec. The pulse-height characteristics of the chamber, given in Fig. 8.5.6, were obtained by using the preamplifier described below. It is evident that gamma pileup presents no problem, even in fields as high as 2×10^6 r/hr. The chamber uses ceramic insulation entirely, and the special cable connecting the chamber to the preamplifier is designed to withstand a gamma-ray dose of 10^{10} r.

Subminiature vacuum tubes are used in the preamplifier rather than solid-state devices because of their better resistance to radiation damage. Each component of the preamplifier will withstand a gamma-ray dose of 10^8 r; this corresponds to at least 1 yr of continuous operation.

The sensitivity of the preamplifier is 3×10^{12} v/coulomb, giving an output pulse height of 150 mv for a fission fragment and 15 mv for an alpha particle. The measured electrical noise at the output is 5 mv peak-to-peak, thus permitting alpha particles and fission fragments to be detected with good signal-to-noise ratios. This design is also optimized to avoid gamma pileup. Each preamplifier receives filament and plate power and chamber polarizing voltage from a power supply (Q-2617) which is supplied from one of the three instrument battery banks. The filaments are supplied from the battery through a voltage regulator, current-limited to protect the filaments. The high voltages for the chamber and preamplifier are supplied by a dc-to-dc converter.

[5]D. P. Roux, *Trans. Am. Nucl. Soc.* 5, 185 (1962).

The output stage of the preamplifier drives one side of a balanced line, the other side of which is terminated symmetrically. This symmetry, together with careful attention to shielding and grounding, makes the assembly, as demonstrated in many field tests, very insensitive to electrical noise pickup.

The receiving end of the balanced line feeds a pulse transformer for common-mode rejection and clipping to give an approximately symmetrical waveform. This avoids base-line shifts due to changes in counting rate and aids in keeping the discrimination level constant.

Pulses from the pulse transformer are next sent to the pulse amplifier and count-rate meter (Q-2614). Following a calibrated attenuator, the pulses are amplified in two feedback stages, each with a gain of 10. They then enter a biased amplifier whose variable bias level determines the discrimination level; only pulses larger than this level are counted. The output pulses from the biased-amplifier discriminator are shaped and applied to a conventional logarithmic counting-rate circuit of the Cooke-Yarborough type. The time constants are chosen as a compromise between the desire to have rapid response and the necessity to prevent excessive fluctuation in the period signal.

A simple linear count-rate circuit supplies the counting-rate signal for the servo which drives the detector.

Built-in oscillators at 10 cps and 10 kc supply calibrating signals. A dc servo amplifier (Q-2615) is used to energize the servomotor for moving the detector.

The function generator is a Vernistat Interpolating Potentiometer, model 2X5 (Perkin-Elmer Corporation) feeding a 34-line function generator. Each of the 34 equally spaced points over the detector withdrawal stroke can be set to within 0.5% of the desired function, correcting as necessary for any departure from exponential attenuation of neutrons with detector withdrawal.

In addition to the wide-range coverage and display provided by this system, a linear signal is also available from a linear count-rate circuit. This linear count-rate signal is displayed and recorded for use of the operator during low-power operation and tests. Several ranges are provided, and provision is made for switching the channel selected for linear display from automatic operation to manual operation so that the chamber position may be determined by the operator. Use of this linear display does not eliminate any of the normal features except that the range is limited if the positioning servo is turned off.

(a) Neutron Source. – The use of a small, relatively insensitive chamber is compatible with the high neutron source level present after the reactor has been in operation for some time. However, to simulate this relatively high source during the initial period of reactor testing would require an unusually large source if the normal startup instrumentation is to be used in the range anticipated for normal operation. For the initial cold startup, additional, more-sensitive detectors were provided. The radioactive source used for the low-power critical experiment program was a 10-curie Po-Be source. The source provided for normal operation is a 64.5-g antimony source located in the reflector.

8.5.5 Miscellaneous Instrumentation

An N^{16} monitoring system is provided in the HFIR for two purposes. First, the N^{16} activity in the exit coolant stream during normal reactor operation should give a good measure of the average reactor power level. Second, the N^{16} system is arranged so that a sample of the coolant is obtained from each quadrant below the core, which should yield information on the azimuthal power distribution. Because sufficient information is not yet available concerning the behavior of such a system to warrant its use in the control system, the present design presents the data to the operator on a multipoint recorder but does not automatically initiate any control action.

The four coolant sample pickup tubes are located at the angular positions of the center lines of the four shim-safety control plates so that each tube samples the water from the region affected by a single plate. The water is picked up at points 21.5 in. below the core center line and is conducted through $^3/_8$-in. tubes to the bottom flange of the reactor pressure vessel. From this point, it is routed through radial passages in the bottom flange and then through tubing to a quadrant selector valve by means of which coolant from any selected quadrant is directed to a sodium iodide scintillation gamma-ray spectrometer.

Work has been in progress at ORNL aimed at developing and demonstrating the techniques necessary for on-line neutron-flux noise spectra measurements. The on-line digital computer has been programmed to perform this function and will be used in further studies to determine the sensitivity to variations in the normal noise pattern such as might be caused by boiling in the lattice, mechanical vibration of core components, or other abnormal conditions. It is planned to incorporate the on-line monitor into the control system if the development program is successful.

8.6 Reactor Process Instrumentation

8.6.1 *Introduction*

This section contains descriptions of the essential process (non-nuclear) instrument complement. As explained previously, certain portions of the HFIR process instrument system have significant control and safety functions and might well have been included in the preceding sections. However, for convenience and because they are non-nuclear in character, they will be considered here. These safety and process instruments include the primary coolant flow-, temperature- and pressure-measuring devices, and the heat-power calculators.

Most of the more conventional process instrumentation is associated with the operation of the various cooling systems. These have been described in considerable detail in Sec. 6, which contains a number of flow diagrams that may be useful in gaining an understanding of the purposes and functioning of much of the process instrumentation.

8.6.2 Process Instrumentation for Safety Channels

The process instrumentation for the safety channels incorporates three separate, completely redundant reactor heat-power computers, utilizing electronic instruments with all solid-state components, which provide (to the nuclear safety instrumentation) signals proportional to reactor heat power, reactor vessel primary coolant inlet temperature, and primary coolant flow rate. The basic transmission signal for these instruments is 10 to 50 ma dc.

In addition, the primary coolant pressure and pony motor flow are each monitored in triplicate and provide signals to the nuclear safety instrumentation.

Testing of each pressure, temperature, and flow sensor (and thus each complete safety channel) can be accomplished during reactor operation and is described in Secs. 8.4.2 and 11.8.

(a) Reactor Heat Power. – Reactor heat power is computed automatically from the primary coolant temperature differential across the reactor vessel and from the primary coolant flow rate. The simplified diagram in Fig. 8.6.1 shows the components making up one safety channel heat-power computer. As noted above, there are three such computers in the safety system.

The differential temperature of the primary coolant is obtained by taking the difference between the reactor inlet (TE 100-1, -2, -3)* and outlet (TE 100-1B, -2B, -3B) temperatures of the primary coolant. The temperature sensors are special, direct-immersion resistance bulbs, with a response time of less than 2 sec. Both inlet and outlet sensors are located in the primary coolant lines as close to the reactor as practicable. To reduce the water transport time from the core to the outlet

*Standard instrumentation nomenclature is followed; Fig. 6.2.2 shows the primary coolant process instrumentation location.

sensors, the outlet line size was reduced from 20 to 18 in., resulting in a transport time of approximately 2 sec at full flow. Both inlet and outlet temperature signals are amplified and converted to signals of 10 to 50 ma dc. Inlet and outlet temperatures for each of the three channels are each continuously recorded in the main control room. In addition, the inlet temperature signals are fed to the nuclear safety system to provide a trip on high inlet temperature. The outlet temperature signals are available for similar use should reactor operating experience later indicate this to be advisable. The differential temperatures are transmitted to the heat-power computers.

The primary coolant flow rate signals are obtained from measurement of the differential pressure developed across a Venturi meter (FE 100). To maintain redundancy of the three separate heat-power computers, the Venturi meter is equipped with three separate sets of piezometer rings with two separate instrument taps on each set of rings. The instrument taps are extended to the special, all-welded, differential-pressure sensors located along the outside of the west wall of the pipe tunnel above the beam room floor. Each 10- to 50-ma dc flow signal is transmitted to the amplifier room, where it is converted to a 10- to 50-ma dc linear signal and is fed to the nuclear instrumentation and to a heat-power computer. Each of the flow rate signals is continuously recorded in the control room.

The heat-power computer receives two separate 10- to 50-ma dc signals: the linearized flow rate and the primary coolant differential temperature. It multiplies them, and multiplies the results by a conversion factor to produce a 10- to 50-ma dc signal directly proportional to reactor heat power in megawatts. Each reactor heat-power signal is continuously recorded in the control room and also is transmitted to the nuclear instrumentation.

(b) Primary Coolant Pressure. – Since the allowable heat flux is a function of primary system pressure (see Sec. 7.5.3), protection must be provided should the pressure fall below an allowable level. It is necessary to shut off electric power to the main coolant pumps in addition to scramming the reactor if a loss-of-pressure accident should occur.

Pressure is monitored by six special, all-welded, Bourdon-tube-type pressure switches. Contacts from three of the switches (PSS 128-A, -B, -C) are provided for the nuclear instrumentation, and contacts from the other three switches (PSS 128-D, -E, -F) are provided for main coolant pump cutoff. The pressure settings for these switches were retained at the original 375 psig for the scram point and 325 psig for pump cutoff; however, the operating pressure at the pressure switches was increased to 650 psi to ensure core inlet pressure of 600 psi.

Each switch can be tested as described in Sec. 11.8.

(c) Low, Low Primary Coolant Flow. – In order to prevent damage to the core due to afterheat following a shutdown, a flow rate of approximately 7% of that at full power must be available immediately after shutdown of the reactor. This flow rate corresponds approximately to that available from one pony-motor-driven pump. Since the ac motors drive the same pumps, it is not possible to apply a test during normal reactor operation which will determine with certainty that they are capable of producing the required flow when required. However, monitoring and testing have been employed to provide the operator with as much information as feasible as to the condition of the pony motors. When the main ac power is lost, the reactor will continue to operate, as will the instrument system and the pony motors, both being supplied with battery power, as described in Secs. 6.2.1 and 8.5.1. The primary coolant flow decreases following loss of electrical power to the primary coolant ac pump motors, until the shutdown coolant flow is reached. Through action of the flow instrumentation, the high flux trip point is continuously adjusted so that an adequate safety margin is maintained. The regulating system maintains the reactor power at a level consistent with the changing flow, making a scram unnecessary. Additional action is necessary following a loss of main power, if operation at about 10 Mw is to continue until such time as the main pump motors can be restarted (i.e., diesel-driven generating units must start). Under these conditions the reactor is immediately ready to be taken back up to the normal operating power level, provided that the duration of the power outage is not long enough to allow excessive accumulation of xenon.

During this period, if actual flow is less than that corresponding to two pony motors, a trip occurs. To accomplish this, a special, all-welded, electric differential-pressure transmitter is paralleled with each heat-power computer flow transmitter. These three low-range transducers are set to trip, through the nuclear safety instrumentation, if the primary coolant flow rate falls below a value approximately halfway between the flow rates provided by two pony motors and by one pony motor. Each motor can be tested as described in Sec. 6.2.1(*d*).

8.6.3 Process Instrumentation for Control Channels

The general philosophy followed in the control and regulation of the reactor is to exercise control through the use of the same variables which, if uncontrolled, could actuate the safety system. This is fundamentally sound provided that, as has been done here, independent channels are provided for the two functions. The process control instruments are therefore duplicates of the safety channel process instrumentation, except they are pneumatically, rather than electrically, activated. As is the case for the safety system, three channels of instrumentation are provided. Reactor differential temperature and heat power from each of three channels are displayed in the control room, and the signal from each of the variables is available to one of the three regulating channels.

The basic signal range of the pneumatic instrumentation is 3 to 15 psig. The instrument air supply system is discussed in Secs. 8.6.17 and 10.3.

(a) Reactor Inlet, Outlet, and Differential Temperature. – The reactor differential temperature is obtained by taking the difference between the inlet (TE 100-4, -5, -6) and outlet (TE 100-4B, -5B, -6B) temperatures of the primary coolant. The temperature sensors are special, direct-immersion, gas-filled bulbs with a response time of less than 2 sec. These sensors are located in the inlet and outlet lines in the same manner as the safety channel temperature sensors.

(b) Primary Coolant Flow Rate. – The primary coolant flow rate signals are obtained from measurement of the differential pressure developed across the same Venturi tube (FE 100) used for the safety channel flow signal. To maintain redundancy, one pneumatic differential-pressure transmitter is piped to one of the two sets of instrument taps provided on each set of piezometer rings of the Venturi meter.

(c) Heat-Power Multiplier. – Each of the three heat-power multipliers receives two separate 3- to 15-psig signals, the linearized flow rate and the reactor differential temperature. These signals are multiplied together, and the result is multiplied by a conversion factor to produce a 3- to 15-psig signal directly proportional to reactor heat power in megawatts.

8.6.4 Primary Coolant Pressure Control

The primary coolant system is pressurized by means of a multistage centrifugal pump (see Fig. 6.2.2). In addition to providing a pressure head for the system, the pump also provides the flow necessary to transfer water from the low-pressure primary coolant cleanup system to the primary coolant high-pressure system. Water is let down from the high-pressure system to the cleanup system through letdown valves (PCV 127-4, etc.) located in each main heat exchanger cell. For a selected operating pressure, there is a corresponding pressurizer pump speed which will provide the desired letdown flow rate. The pressurizer pumps have variable speed drives so that the desired speed may be obtained by remote manual adjustment. The letdown valves are controlled by a pressure-sensing element (PT 127), which adjusts the letdown valves to maintain constant system pressure.

(a) Pressurizer Pumps. – Two multistage, variable-speed centrifugal pumps are provided, one of which is for "standby" duty. A third centrifugal pressurizer pump with sufficient capacity to offset system losses and maintain circulation pump seal flow is installed to maintain system pressure during a power failure. This pump is operated by the normal-emergency ac power supply.

Pump motor current indicators, "running" lights, speed indicator, manual speed controller, and manually operated electric switches for "start-stop" and "standby" control of the pump motors are located in the control room for each of the two main pressurizer pumps. The standby pump will automatically start on loss of pressure in the primary system if normal-power is available. The emergency pressurizer pump is automatically started on low-flow indication in the main pressurizer pump discharge lines.

(b) Letdown Valves. – After the nominal 200-gpm letdown flow to the primary coolant cleanup system has been established by adjusting the pressurizer pump speed, primary system pressure is controlled by three (one from each operating pump–heat exchanger cell) bellows-sealed, pneumatically actuated control valves (PCV 127-4, etc.) which "let down" primary high-pressure water to the low-pressure cleanup system. For good flow control near the closed position, the valves are equipped with plugs and seats having "equal percentage" flow characteristics. These valves are controlled by a special bellows-type, all weld-sealed, pneumatic, pressure transmitter-controller. Primary coolant pressure is recorded continuously, and off-limit pressures are annunciated in the control room. In addition, in each letdown line, two pneumatically actuated, spring-loaded-closed block valves (PCV 127-1A, etc.) are installed, in series, downstream of the flow control valves and will automatically close in the event of gross loss of pressure in the primary high-pressure system or on indication of a gross increase in activity in the primary coolant system. These block valves are closed by a separate Bourdon-tube pressure switch below 550 psig.

8.6.5 Primary Coolant Temperature Control

The temperature of the water leaving the primary heat exchangers is controlled by regulating the temperature and flow of the secondary coolant water to these exchangers. The secondary water which enters the tower basin after passing through the cooling tower is maintained at an approximately constant temperature by automatic selection of the number and speed of the operating cooling tower fans. The secondary coolant flow to the primary heat exchangers is controlled by a 10-in. throttling valve (TCV 377A) around the main 36-in. valve (TCV 377) in the common header line to the heat exchangers. Through the action of this valve, the temperature of the primary system water leaving the heat exchangers may be held constant despite variations in reactor power. Manual control of this valve is provided in the control room in addition to automatic control based on maintaining the desired reactor inlet temperature. The large valve is adjusted to keep the smaller valve in midrange.

8.6.6 Primary Coolant Circulation Pumps

Four centrifugal pumps are provided to maintain primary coolant circulation. Each pump is connected to a primary coolant heat exchanger in a separate shielded cell. During normal operation, three pump–heat exchanger combinations are operating, with the fourth unit in standby.

During a power outage, the primary coolant flow necessary for afterheat removal is maintained by a battery-powered pony motor coupled directly to the shaft of each pump. A separate battery bank and battery charger are provided for each pony motor.

Main motor current indicators, pony motor current indicators, "running" lights, and manually operated electric switches for "start-stop" control of the main pump motors are located in the control room. Both high and low pony motor currents are annunciated in the control room.

Gate-type block valves are installed in the suction and discharge of each pump–heat exchanger combination. Actuating switches and "open-closed" indicating lights for each valve are located in the control room. Each suction valve is provided with a small, pneumatically actuated, bellows-sealed globe valve which is opened before startup of a pump–heat exchanger combination to allow equalization of pressure across each gate valve in order to minimize gate wear during the opening cycle. An actuating switch and "open-closed" indicating lights for this valve also are located in in the control room.

8.6.7 Miscellaneous Primary System High-Pressure Instrumentation

(a) Reactor and Inlet Strainer Differential Pressure. – The primary coolant differential pressures across the reactor (PdT 106) and the inlet strainer (PdT 103) are monitored by special weld-sealed, pneumatic differential-pressure transmitters. The reactor and the strainer differential pressures are recorded in the control room, and both reactor high and low differential pressures are annunciated.

(b) Temperature Measurements. – Primary coolant temperatures are measured at the common inlet of the heat exchangers, at the outlet of each heat exchanger, and at the discharge header of the pressurizer pumps; all are indicated in the control room.

(c) Reactor Vessel Venting. – A pneumatically actuated, bellows-sealed valve is provided to vent the gases accumulated in the flange penetrations and the top of the vessel during reactor shutdown and operation. The line from the vent valve leads to the low-pressure letdown header. "Open-closed" lights and an actuating switch are located in the control room. The vent valve is spring-loaded to fail closed.

(d) Pressure Gages. – At the inlet and outlet of each of the four primary coolant circulation pumps and of each of the three pressurizer pumps, locally mounted pressure gages are installed to facilitate maintenance and testing.

8.6.8 Primary Coolant Cleanup System

(a) Deaerator. – Primary coolant from the letdown header is directed through a vacuum deaerator for removal of entrained and dissolved gases (see Fig. 6.2.11). Deaerator vacuum is indicated and annunciated in the control room. Deaerator water level also is controlled, with indication and annunciation of high and low levels in the control room. FE 917 monitors dilution air admitted to the CHOG system at the condenser, and low flow is annunciated.

A high high-level switch is provided which will close the block valve in the steam supply line to the deaerator gas ejectors in the event the deaerator level becomes uncontrolled and the deaerator is in danger of flooding. Similarly, a high-temperature switch will close the block valve in the steam supply in the event that steam-jet condenser water fails, thus preventing steam from reaching the off-gas filters. This high high-level condition is annunciated in the control room.

(b) Cleanup Pumps. – Two horizontal-type centrifugal pumps, one normally operating and the other on standby, provide circulation through the primary cleanup system. Pressure gages are installed at the inlet and outlet of each pump for maintenance and testing operations, and a dial-type thermometer is installed in the common discharge header.

Pump "running" lights and manually operated electric switches for "start-stop" control of the pump motors are located in the control room.

Conductivity and pH of the pump discharge stream are measured, indicated locally, and continuously recorded in the control room.

(c) Filter and Demineralizers. – An inlet filter and a demineralizer, made up of cation and anion beds, are furnished for primary coolant cleanup. Pressure gages to monitor the pressure drop and to help detect plugging are installed at the inlets to the filter, at the cation and anion beds, and at the outlet of the demineralizer system.

Conductivity and pH of the exit water stream from the demineralizer after-filter is monitored continuously and is locally indicated, and recorded in the control room.

Sample lines from the inlet and outlet of the filters and of each cation and anion bed are brought through the biological shielding to a sink to permit sampling for pH, conductivity, and radioactivity analyses. Each line is provided with a solenoid valve which automatically closes on a high-radiation signal from a monitron near the sink. The sink radiation level is indicated and annunciated locally and also is annunciated in the control room. A radiation detector, using a nonsaturating Geiger tube, monitors the demineralizer effluent for gross low-level activity. This measurement is locally indicated and is annunciated in the control room.

(d) Head Tank. – The cleanup system discharges into a head tank which supplies the pressurizer pumps and which is vented to the open hot off-gas (OHOG) system. The level in the head tank is monitored by a pneumatic level-controller which operates a control valve in the demineralized water makeup line. Low water level shuts off the main pressurizer pumps. The level is indicated and annunciated in the control room. A locally mounted flow totalizer is installed in the makeup line.

(e) pH Control System. – ApHE 1200 is provided to control the rate of nitric acid addition to the head tank necessary to maintain a pH of ~5.0 in the primary coolant systems; however, manual control has proved to be satisfactory.

8.6.9 Secondary Coolant System

(a) Cooling Tower. – There are four individual cooling tower cells assembled as a unit on a concrete basin. The basin is divided into four compartments. The compartments lead to a flume, which feeds the suction well for the secondary coolant pumps. Clear space and blanked piping connections are provided for a future fifth cell (see Fig. 6.3.2).

The liquid level in the flume is measured and maintained by a local controller which supervises the flow of makeup water required to offset evaporation and blowdown losses. Abnormal liquid level is annunciated in the control room by a level switch that is also mounted in the flume.

The basin blowdown flow rate is automatically controlled by a pneumatic cascade-control system (i.e., the makeup water flow rate measurement is pneumatically transmitted to adjust the setpoint of the pneumatic blowdown flow controller in order to maintain a predetermined blowdown-to-makeup flow ratio). Makeup and blowdown flow rates are each indicated locally, and a radioactivity detector monitors the blowdown line. The radioactivity measurement is recorded.

There are eight cooling tower fans (two per tower) with two-speed motor drives utilizing automatic sequential "on-off" control to maintain a preset temperature of the water falling into the tower basin. The fans are controlled by one temperature controller. The temperature measurement is recorded and annunciated in the control room. Pressure in the discharge manifold of the cooling tower pumps is electrically transmitted and is indicated in the control room.

The addition of sulfuric acid to the cooling tower basin is automatically controlled to maintain a selected pH value in the discharge manifold of the cooling tower pumps. The pH measurement is locally indicated and controlled, and a remote record is provided in the control room. The rate of flow of the makeup water used in the chemical-addition process is locally indicated.

(b) Secondary Coolant Pumps. – Three vertical-type turbine pumps connected to the normal ac power supply provide circulation for the secondary coolant system. An auxiliary pump with two windings is provided. One winding, supplied from normal-power, provides circulation during shutdown; the other, supplied from normal-emergency power, provides circulation for operation at N_L during a normal-power outage (see Sec. 6.5). A bypass line is provided on the cooling tower distribution piping to permit return of the secondary coolant water directly to the tower basin during operation on emergency power. A block valve is provided in this line to open only when the auxiliary pump is operating on emergency power, to reduce the electrical power required to run the pump. The cooling tower basin has adequate heat capacity to allow this method of operation for an extended period of time. Locally mounted pressure gages are installed in the outlet of each pump for use in maintenance and testing operations.

Pump "running" lights and manually operated electric switches for remote "start-stop" control of pump motors are located in the control room. Pump motor current is locally indicated in the electrical equipment building.

(c) Secondary Coolant to Primary Heat Exchangers. – Thermometer wells only are provided at inlet and outlet of secondary coolant piping to the primary heat exchangers to facilitate acceptance testing of the heat exchangers. In addition, the common inlet temperature of the heat exchangers and the outlet temperature of each heat exchanger is measured with thermocouples and indicated in the control room.

The differential pressure across the secondary coolant common inlet and outlet to all primary heat exchangers is indicated in the control room, and pressure gages are installed in the inlet and outlet of each heat exchanger.

The total flow of secondary coolant through the primary heat exchangers is determined from the measurement of differential pressure developed across a Dahl flow tube located in the inlet manifold. The flow rate is transmitted electrically for control room indication.

Remotely operated block valves are installed in the secondary coolant inlet and outlet of each primary heat exchanger to permit optional circulation of secondary coolant through any combination of the individual exchangers. Position-indicating lights and manual switches are located in the control room for "on-off" operation of these valves.

A flow control valve (TCV 377A) regulates the flow to all heat exchangers in order to obtain temperature control of the primary coolant leaving the heat exchangers. This valve is provided with a manual loading station in the control room in addition to the automatic control feature.

(d) Secondary Coolant to the Pool Coolant Heat Exchanger. – Secondary coolant flow rate is measured at the outlet of the pool heat exchanger bank and is indicated locally; locally mounted differential-pressure indicators are installed across the secondary coolant inlet and outlet of each pool heat exchanger.

The temperature of the pool water at the outlet of the pool heat exchanger bank is measured and maintained at a predetermined setpoint by automatically controlling the flow of secondary coolant through the exchangers. The measured temperature is locally indicated.

The pH of the secondary coolant to the pool heat exchanger is measured and locally indicated. This signal is recorded in the control room, and high and low pH values are annunciated. The pH system is partly redundant since it is also measured by the system provided for acid-addition control described in Sec. 8.6.9(*a*); however, this system provides a check on the operation of the acid-addition-control system.

8.6.10 Pool Coolant System

(a) Pools. – A coolant system is provided for the reactor pool, the clean pool, and the critical pool (see Fig. 6.4.2). The flow to the reactor, clean, and critical pools is measured and is indicated in the control room. The temperatures of the inlet and outlet of the reactor and critical pools are measured and indicated in the control room. The combined scupper drain from the reactor, clean, and critical pools is monitored for radioactivity. The measurement is locally indicated and is recorded and annunciated in the control room. The drain discharges into the pool surge tank.

The level of each pool is monitored, and both the high and low levels are annunciated in the control room. A high-level signal shuts down the pool coolant circulation pumps.

(b) Surge Tank. – The pool surge tank is vented to the OHOG system and drained to the process-waste system (see Sec. 11.3.2).

Liquid level in the surge tank is automatically controlled at a selected height by throttling the demineralized-water makeup stream. The pneumatic level controller is locally mounted, and the measurement is indicated and annunciated in the control room. A low-level signal shuts down the pool coolant pumps.

(c) Pool Coolant Circulation Pump. – Two horizontal-type centrifugal pumps, one normally operating and the other on standby, provide circulation during normal operation of the pool coolant loop.

Pump "running" lights and manually operated electric switches for remote "start-stop" control of pump motors are located in the control room, and locally mounted pressure gages are installed in the inlet and outlet of each pump for use in maintenance and testing operations.

(d) Filter and Heat Exchanger. – Locally mounted pressure gages are installed in the inlet and outlet of the pool coolant filter and in the pool coolant heat exchanger discharge header. The reading of the filter bank outlet pressure gage also serves as the pressure measurement of the heat exchanger inlet.

Temperature indicators, also locally mounted, are installed in the inlet of the pool coolant heat exchangers as well as in the outlet of each exchanger. The reading of the exchanger outlet header temperature sensor is indicated in the control room and serves as the temperature measurement of the combined coolant inlet to the reactor and critical pools.

A remotely controlled, electrically operated block valve is installed in the individual "drain and fill" line of the reactor, clean, and critical pools. Normally these valves are maintained in a fully open position; however, they may be set to produce a desired variation in coolant flow to the several pools. Position-indicating lights and manual switches are located in the control room for operation of the valves.

8.6.11 Pool Cleanup System

(a) Defective Fuel Element Storage. – Reactor pool water from the defective fuel element storage tanks located in the reactor pool is monitored for radioactivity, and the measurement is locally indicated as well as recorded and annunciated in the control room (see Fig. 6.4.5).

Temperature and flow of the water in the defective fuel element tank outlet line are measured and indicated locally in the control room.

A remotely controlled, electrically operated block valve is installed in the defective fuel element tank outlet line from the reactor pool. Normally this valve is maintained in a fully open position; however, it may be closed from the control room if necessary in the event of a water leak or for other reasons. Position-indicating lights and a manual switch are located in the control room for operation of this valve.

(b) Pool Cleanup Deaerator. – Pool water from the defective fuel element tank outlet is directed through a vacuum deaerator for removal of entrained or dissolved gases. Accumulated gas is drawn off by a steam-ejector system. Intermediate- and after-condensers are provided in the deaerator gas discharge line for removal of the condensables before the gas is sent to the closed hot off-gas (CHOG) system. Deaerator vacuum is indicated in the control room, and a high-pressure condition is annunciated.

Pressure in the intermediate- and after-condensers is locally indicated, and the liquid in the bottom of the deaerator is automatically controlled at a selected height by throttling the discharge of the pool demineralizer pumps. The pneumatic level controller is locally mounted, and the measurement is remotely indicated and annunciated in the control room. Interlocks stop the pool demineralizer pumps on low level and restart them on level rise. High, high level in the deaerator is annunciated in the control room, shutting off steam to the ejector. High gas discharge temperature also shuts off steam, to prevent steam from reaching the off-gas filters should condenser cooling water fail.

(c) Pool Demineralizer Pumps. – Two horizontal-type centrifugal pumps, one normally operating and the other on standby, provide circulation through the pool demineralizers.

Locally mounted pressure gages are installed in the inlet and outlet of each pump for use in maintenance and testing operations. Pump "running" lights and manually operated electric switches for remote "start-stop" control of the pump motors are located in the control room. Interlocks for starting and stopping the pump on normal and low deaerator water level are provided.

(d) Pool Demineralizer. – The pool demineralizer consists of a cation unit and an anion unit in series. Flow to the demineralizer is measured and locally indicated. Locally mounted pressure gages are installed in the inlet and outlet of the cation unit as well as in the inlet to the anion unit. Sample lines from the inlet and outlet of the anion and cation beds are brought through the biological shielding to a sink for manual checking of pH and for conductivity and activity analyses. The sink radiation level is monitored and locally indicated and annunciated; it is also annunciated in the control room.

(e) Return Lines. – Pool demineralizer effluent discharges through an afterfilter and feeds the clean pools; the flow is measured and indicated in the control room. The pH and conductivity of the water leaving the afterfilter are monitored and locally indicated. Water from the pool cleanup system joins the pool coolant flow just upstream of the block valves noted in Sec. 8.6.10(*d*).

8.6.12 Reactor Primary Coolant and Pool Water Storage Tanks

A buried concrete water-storage tank with two separate compartments is provided for draining the reactor primary coolant loop and the pool coolant loop. The liquid level in each compartment is measured and indicated in the control room.

8.6.13 Demineralizer Regeneration

The two demineralizer units, one serving the reactor primary coolant loop and one serving the pool coolant loop, are designed for in-place regeneration, with provision for backwash and regeneration of each bed. Locally mounted pressure gages are installed in the suction and discharge of the recycle pumps for use in maintenance and testing operations.

Regeneration is manually controlled from a centrally located instrument panel in the operating area adjacent to the demineralizer units. Transfer of damaged resin to disposal containers for subsequent burial also is controlled from this panel (see Sec. 11.4).

8.6.14 Plant Process Water Supply

Process water is supplied to the demineralizers and other process water users by two horizontal-type centrifugal pumps, one normally operating and the other on standby (see Fig. 10.2.2). Locally mounted pressure gages are installed in the inlet and outlet of each pump for use in maintenance and testing operations. Pump "running" lights, a pressure indicator, and manual switches for remote "standby-start-stop" control of the pump motors are located in the control room.

8.6.15 Plant Demineralized-Water System

The plant water demineralizer is a packaged monobed unit (see Fig. 10.2.2). Flow of water to the demineralizer bank is measured and locally indicated. A locally indicating flow totalizer is installed in the demineralizer outlet. Locally mounted pressure gages are installed in the inlet and outlet of the demineralizer. Conductivity of the plant demineralizer effluent is locally indicated, as well as in the control room.

In-place regeneration is employed, with provisions for backwash and regeneration of each resin. Regeneration is manually controlled from an instrument panel in the operating area adjacent to the demineralizer system.

A demineralized-water storage tank is provided for filling the reactor vessel and coolant loop and as a source of normal makeup for both the reactor and pools systems. The tank is maintained full by controlling the makeup flow. High and low levels are annunciated in the control room. Demineralized water is supplied to the reactor primary coolant system and the pool coolant system by two horizontal-type centrifugal pumps that take suction from the demineralized-water storage tank. Normally, one pump is operating and the other pump is on standby. Locally mounted pressure gages are installed in the inlet and outlet of each pump for use in maintenance and testing. Pump "running" lights, a pressure indicator, and manually operated switches for remote "start-stop" control of the pump motors are located in the control room.

8.6.16 Caustic and Nitric Acid Supply Systems

Strong caustic and nitric acid are drawn from the storage tanks by gravity and diluted in day tanks to the required concentrations by the addition of demineralized water. The weak solutions are stored in the day tanks and distributed to the filters and demineralizers on demand. The liquid levels in the day tanks are indicated locally. The temperatures in the caustic storage tank and the day tank are controlled at a selected value by throttling steam flow to heating coils.

Centrifugal pumps are used for circulating the diluted solutions to the users. Locally mounted indicating pressure gages are installed in the suction and discharge of each pump for testing and maintenance purposes. "Running" lights and manually operated electric switches for "start-stop" control of the pumps are mounted on the local control panel.

8.6.17 Instrument Air System

An instrument air supply system is provided for pneumatic instrument application. Figure 8.6.2 is a simplified block diagram of this system; a more detailed diagram appears in Fig. 10.3.1. Three compressors are furnished, each capable of delivering 100 scfm of completely oil-free air at pressures of 60 to 70 psig. Each of these compressors is supplied power from separate normal-emergency systems. Normally, one compressor is operating and two are on standby. Air supply pressure is indicated in the control room, and a low-pressure condition is annunciated. A control device is provided to start the standby compressors if the air supply pressure falls to 53 psig. Once started, the standby compressors remain in operation until manually switched off. "Running" lights and hand-operated switches for "start-standby-stop" control of the compressor motors also are provided in the control room.

Two complete "heatless" drying and filtering systems are provided, each rated at 200 scfm of dry air with a dew point not higher than −20°F. Recycling of the drying and filtering systems is automatic. Since recycling of a dryer system consumes 50 scfm of air; the net rating of each is 150 scfm.

Air is distributed at a pressure of 60 to 70 psig and, in general, is reduced at the instruments as required by means of manifolds, each supplying several instruments. Pressure-reducing stations for these manifolds consist of a parallel arrangement of two filters and two pressure regulators. Block valves are provided to facilitate component replacement.

An emergency air header is provided to assure air supply to certain instruments. This header is normally supplied with air through a check valve from the instrument air compressors; however, if the normal compressors fail to supply adequate air or if some other part of the normal system fails, the emergency air compressor can supply air to this header. The emergency compressor is also supplied with power from one of the normal-emergency systems. The emergency header supplies air to the three regulating channels, each of which is isolated from the emergency header by a check valve and is provided with a capacity tank with sufficient supply for 10 min of operation. This header also supplies air to the letdown control and block valves and the secondary coolant throttling valves TCV-377 and 377A. The emergency compressor automatically starts when the pressure in the emergency header falls to 43 psig. Once started, the emergency compressor will remain in service until manually shut down.

8.7 Radioactivity Monitoring

8.7.1 Introduction

There are four radioactivity monitoring systems in the HFIR complex: the Health Physics Monitoring System (HPMS), the Gaseous Waste Monitoring System (GWMS), the Liquid Waste Monitoring System (LWMS), and the Coolant Activity Monitoring System (CAMS).

The HPMS is provided primarily to ensure that the radiation level in normally occupied areas is sufficiently low to permit unrestricted access for operating and service personnel. This system, by means of control room alarms, makes known situations requiring attention of the operating staff and, by means of local alarms, indicates to personnel in the affected area that the radiation has exceeded the normal level. This network is also a part of the ORNL Radiation Warning and Communication System.[6] Instruments at selected locations in the building are connected to the HFIR evacuation alarm and are arranged so that a coincidence of two high-level alarms from the selected constant air monitors or a coincidence of two alarms from the selected monitrons will automatically set off the evacuation horn for the entire facility. Smaller or more localized incidents will be handled from the control room by the operator or supervisor. In addition to the local evacuation signal, the system transmits signals to the ORNL Emergency Control Center, which also receives information from other ORNL facilities. This center also receives information from the ORNL Waste Monitoring Control Center and the Health Physics Environmental Monitoring Control Center.[7] In the event of an incident, appropriate action can be taken at the Emergency Control Center.

---

[6] *Radiation Safety and Control at the Oak Ridge National Laboratory, 1960–1962,* ORNL-TM-507 (Apr. 5, 1963).

[7] *Applied Health Physics Annual Report for 1962,* ORNL-3490, p. 89 (Sept. 25, 1963).

The GWMS is provided to monitor the stack effluent, and alarm and indication are provided in the control room. This system also transmits information to the ORNL Waste Monitoring Control Center.

The LWMS is provided to monitor the several liquid-waste lines which serve the facility. By means of alarm and indication in the control room, information on the activity level is made available to the operator. In addition, the monitors actuate diversion valves which can direct normally uncontaminated waste to holding ponds should the activity level increase to a point beyond normal. This system also transmits information to the ORNL Waste Monitoring Control Center.

The CAMS is provided so that the operator may be quickly informed of a significant increase in the activity of the various coolant systems and so that he will have adequate information to help identify the source of the trouble. Information from this system is annunciated and recorded in the control room.

As an aid to analysis of a serious accident, should one occur, there are five threshold detector units for nuclear incident dosimetry located so that each can be retrieved from outside the reactor building.

8.7.2 Health Physics Monitoring

The installed HPMS is provided to give information on the radiation and air-activity levels in normally unrestricted access areas of the building. This system is made up of a constant air monitor system and a radiation monitoring system utilizing beta- and gamma-sensitive monitrons. In addition to the instruments utilized for general building coverage, a number of Geiger-Mueller (GM) tube monitors are located about the building for radiation measurements during operation and maintenance. A hand-and-foot counter is provided for contamination checks, and a variety of portable instruments are available for specialized monitoring.

(a) Particulate and Gas Activity Monitors. – Air in potentially contaminated regions of the HFIR building is continuously monitored for airborne radioactive material, whether particulate or gaseous in nature. The most likely potential sources of activity are leakage from the primary coolant system or a release from fuel elements or targets either in the reactor or in the storage areas of the pools. The monitoring is accomplished by passing air through filter-paper tape and monitoring the collected material by means of a GM tube. The air, after passing through the filter, flows around the GM tube; therefore radioactive gases are also monitored. Since the GM tube is shielded from direct radiation, the observed activity level is the result of either airborne radioactive material or an extremely high level of direct radiation. The latter, however, would have been detected at a much lower level by other unshielded radiation monitors. Standard Constant Air Monitors (Q-2240) are used, and alarms are interpreted according to standard health physics procedures.

Nine constant air monitors are located so as to provide coverage of the operating areas of the building. They are located as follows:

Subpile room	1
Ground floor	
Water wing	2
Beam room	1
First Floor	
Water wing	1
Experiment room	1
Second floor	
Reactor bay	2
Observation gallery	1

All of the constant air monitors will give a local intermediate- and high-level alarm; and all units also transmit information to the control room, where alarm levels are indicated and activity levels are recorded on panel H.

(b) Radiation Monitors. – The monitoring system for direct beta and gamma radiation is similar in arrangement to that of the constant air monitoring system. Nine ion-chamber-type monitrons (Q-1154) are provided and their locations are as follows:

Subpile room	1
Ground floor	
Beam room	1
First floor	
Experiment room	1
Water wing	1
Second floor	
Reactor bay	4
Observation gallery	1

Provision is made for the addition of eight more units should they be needed to provide monitoring around experiment installations.

The monitrons have local intermediate- and high-level alarms, and information is transmitted to the control room for annunciation and indication.

A special high-range radiation monitor is located in the reactor bay beneath the observation gallery windows. Two readouts (.1 to 10^5 r/hr) are provided, one in the control room and one in the office and maintenance building. A 5 r/hr alarm will ring in the control room.

(c) Miscellaneous. – Several other types of health physics monitoring equipment are provided. Laboratory monitors of the GM-probe and count-rate-meter type are installed in locations where there may be need for this type of instrument. The laboratory monitors have a local alarm which will be set at 2.5 mrems/hr. In addition to their use as probe-type instruments, the local alarm feature will make known abnormal radiation levels to persons in the immediate vicinity.

A hand-and-foot counter is provided for routine checks on personnel contamination.

The sampling sinks in the process water area are provided with radiation monitors to minimize the chance of an operator receiving an excessive radiation dose during sampling operations. In addition to actuating an alarm, the instruments will also close solenoid valves in the sampling lines on an abnormal radiation level.

The usual portable instruments (such as ion chamber monitors, GM-tube survey meters, thermal- and fast-neutron dosimeters, pocket dosimeters, and high-volume air samplers) are provided. Beta-gamma and alpha smear counters are installed in the Health Physics Room (206). Special instruments, if needed, are available from the Health Physics Division. A 128-channel gamma analyzer is provided for various analyses.

8.7.3 Gaseous-Waste Monitoring

Three separate air or gas effluent systems from the HFIR building combine to discharge through the stack. The special building hot exhaust system (SBHE) is a high-flow, normally low-activity system which handles all ventilation exhaust air from the containment portion of the building. The airflow in this system is normally about 29,000 cfm. Two other systems, the open hot off-gas system (OHOG) and the closed hot off-gas system (CHOG), are designed to handle normally contaminated air and gas from the coolant system deaerators, storage tanks, etc. The OHOG system is designed for miscellaneous venting in which the source of radioactive gas is not pressurized; and, therefore, if vacuum in the OHOG system is lost, the radioactive gas from one venting location will not be transmitted to another location through the OHOG system. The CHOG system is a completely closed system in which gaseous discharge from devices such as a deaerator may continue, even if off-gas vacuum is lost (see Sec. 4 for descriptions of the systems).

Exhausts from the three systems (SBHE, OHOG, and CHOG) leave the building and pass underground to separate filter banks, as described in Sec. 4.7. The blowers are located after the filter banks. The HOG blower exhausts are combined in a common duct which leads to the inlet of the SBHE filters. The SBHE blowers discharge into a common duct leading to the base of the stack.

A platform encircling the stack is provided at the 50-ft level for instruments, and penetrations are provided for taking samples of the stack air stream at this level.

Under normal conditions the SBHE system contributes essentially no activity to the stack effluent. The majority of the radioactive material comes from the CHOG system and originates in the primary coolant deaerator. The radiation will be from predominantly mixed fission product gases which have their origin in the U^{235} contamination on the fuel element surfaces and contamination which may result from tramp uranium in the system from previous cladding failures. Since the reactor target loading will contain gram quantities of the heavy elements, an alpha monitor is included in addition to the normal beta-gamma monitors. This appears to be unnecessary as far as the HFIR is concerned since any alpha-active material would be accompanied by the beta- and gamma-active fission products; however, use of this stack by the Transuranium Processing Plant (TRU) makes installation of the alpha detectors advisable. A continuous monitor for iodine is also provided.

Details of the gaseous-waste monitoring system are given in Fig. 8.7.1. Figures 4.5.1 and 4.6.1 are schematic flow diagrams of the gaseous-waste systems and show the location of the several monitoring points.

(a) Stack Monitoring Channels. – Monitoring of exit air is accomplished in seven channels as follows:

Beta-Gamma Particulate and Gas Activity Stack Monitor. – This channel (RE-907) utilizes a beta-gamma particulate monitor based on the ORNL Q-2240 air monitor. Particulate matter is collected on a section of filter-paper tape from a stream of about 3 cfm withdrawn from the stack flow at the 50-ft level. The collected sample is monitored by a thin-window GM tube in a stainless steel shield. Only the shielded GM tube, the filter-paper tape, and the vacuum pump are located at the 50-ft instrument platform. Vacuum pump operation is monitored by a pressure switch, and filter tape break is monitored by microswitches in the tape advance unit. It is possible to advance the filter tape either from the 50-ft level or from the amplifier and relay room, where the electronic equipment for this monitor is located.

Alpha Activity Stack Monitor. – An arrangement similar to that described above, but utilizing an alpha particulate monitor (RE-908) based on the ORNL Q-2340 air monitor, is provided at the 50-ft level for monitoring alpha activity.

Beta-Gamma Particulate and Gaseous Activity Duct Monitor. – A second beta-gamma particulate and gas activity monitor (RE-912) is located on the ductwork at the foot of the stack and also reads out in the amplifier and relay room. This monitor samples the mixed effluent air from the HFIR duct at the entrance to the stack.

Iodine Monitor. – A side stream of stack gas is withdrawn and passed through an activated-charcoal trap in the iodine monitor, where iodine and other similar gases are absorbed on the charcoal. The gas sample is returned to the inert-gas monitor after passing through the charcoal. The detector assembly is a ring of GM counters surrounding the charcoal filter. The trap and detector assembly are enclosed in a lead shield to reduce background radiation. The GM tubes feed a count rate meter, an alarm and a recorder. The trap is easily replaceable and will be replaced when the count rate reaches an operationally determined limit. The monitor is called RE-910.

Inert-Gas Monitor. – The stack gas leaving the iodine monitor passes through a charcoal filter, then through a small lead-shielded tank, and then back to the stack. An end-window G-M tube (RE 911) in the tank monitors $\beta-\gamma$ radiation emitted from inert gases which pass through the particulate and charcoal filters.

In-Stack Monitor. – An in-stack monitor is provided which consists of a small quantity of charcoal and a filter-paper disk assembled as a cartridge. This cartridge is located directly in the stack air stream at the 50-ft level, with a separate pump pulling a sample of stack gas through the cartridge at all times. The cartridge may be withdrawn and inserted from the instrument platform. Pump operation is monitored by a pressure switch.

High-Level Stack Monitor. – An ion chamber (RE 909) is located in a thimble placed in the stack, to provide a gross gamma indication of stack effluent. A remotely actuated carriage varies the position of a 1-mc Cs^{137} source to check the operation of the system. The instrument has a control room indicator and is sensitive from 0.025 to 2.5×10^6 r/hr.

(b) Intended Operation. – The particulate monitors at the 50-ft level are intended to give alarms due to bursts or increased levels of gaseous and particulate matter in the stack air stream. The alarm will be annunciated and the level recorded in the control room. The filter tape can be advanced to determine if the alarm was valid or was from an instrument malfunction. The tape advance can also help determine whether the activity is gaseous or particulate. The alpha channel functions in an identical manner except that the detector is sensitive to alpha emitters only. The beta-gamma monitor installed on the ductwork at the base of the stack will help identify the source of radioactive material in the stack when there are other contributors to the stack stream such as the TRU facility. Its operation is identical with the other two stack alarm channels. The iodine monitor will alarm on an abnormal increase in activity trapped on the charcoal. The in-stack sampler will provide a record of average stack concentrations and aid in determining the type of activity discharged. The cartridge may be withdrawn from the stack stream and analyzed in the laboratory.

The inert-gas monitor indicates increases in fission gas or activated air. The high-level gross gamma is intended as a check should other, low-level, instrumentation suddenly indicate at or near full scale.

In addition to the beta-gamma monitor located in the ductwork leading to the stack, provisions have been made for sampling the inlet and exit air from each filter bank in each of the three exhaust systems. Portable samplers may be attached for measurement of either particulate or gaseous activity, or gas samples may be passed through a cartridge for laboratory analyses. These provisions may help to identify which of the systems is releasing activity and allows for routine testing of the filter system.

(c) ORNL Waste Monitoring Control Center. – A Central Waste Monitoring Control Center is operated by the ORNL Operations Division. This facility is organized to monitor essentially all gaseous and liquid waste systems in the ORNL area. Information from the various systems is transmitted to this Control Center, where the data is displayed. This central monitoring system is provided in order to allow rapid identification of sources of activity and to provide information which will allow prompt action in handling routine and emergency activity releases.

8.7.4 Liquid-Waste Monitoring

Potentially contaminated liquid waste from the HFIR is handled by three systems (described in Sec. 11.3) which are separated on the basis of potential contamination level. Figure 8.7.2 is a schematic flow diagram of the waste system showing the functional location of the monitoring and control points. They are: (1) the cooling tower blowdown system, which normally has no contamination; (2) the process-waste water system (PWD), which is normally uncontaminated or only slightly contaminated, but which can be subject to contamination; and (3) the intermediate-level-waste system (ILW), which will always be contaminated to some extent. The process-waste water system collects water from floor drains, experiment stations, equipment drains, and equipment drain pads. The sources of ILW waste are all in process areas in the reactor building and result from primary coolant leakage, demineralizer regeneration fluids, hot sink drainage, etc. Several routes are provided to allow the handling of water of various levels of contamination. The cooling tower blowdown is allowed to enter Melton Branch directly, provided it is uncontaminated. Two ponds of 240,000 and 500,000 gal capacity are provided for holding water before its release to Melton Branch. If the water is found to be contaminated, it will be transferred to the ORNL Central Waste Collection System. A 13,000-gal stainless steel holding tank is provided for special handling of the contaminated ILW water. The normal route for discharge from the ILW system is to the ORNL Central Waste Collection System,[8] in which the waste will be analyzed and handled according to the type and activity level of the contamination. The contaminants in the liquid waste from the HFIR may contain alpha, beta, and gamma activity; however, the probability of having alpha activity without associated beta-gamma activity is very small.

[8] F. L. Culler (internal memorandum), *Criteria for Handling Melton Valley Radioactive Wastes* (May 1961).

(a) Cooling Tower Blowdown Monitor. – Blowdown from the cooling tower is automatically controlled to maintain a preset ratio to the amount of makeup water added. The blowdown line comes from the distribution header on the inlet to the tower. The water in this header is continuously monitored for low-level activity, as described in Sec. 8.7.5, in order to allow early detection of a heat exchanger leak; the unit will function whether or not there is any blowdown flow.

The blowdown flow passes from the cooling tower distribution header to the blowdown flow control valve, through the liquid-waste monitoring station No. 1, and directly to the Melton Branch of White Oak Creek if not contaminated. It may be diverted to the No. 1 holding pond if a high activity level is detected.

Monitor station No. 1 includes a flow measurement system, a sample collection system, and a radiation monitoring system. The flow measurement system transmits information to a flow recorder located in the control room. In addition, the flow instrumentation integrates the flow rate to obtain total flow and also provides a flow-proportional signal for operation of the proportional-sampling system. The collected sample is contained in a plastic bottle; laboratory analyses can be made to determine the type and quantity of activity.

The monitor also measures the radioactivity of a sample stream using an immersed GM tube for β-γ, and a zinc sulfide crystal for α. Information is recorded in the control room, and alarms are provided to bring abnormal conditions to the operator's attention. The alarms also actuate diversion valves which will automatically send the blowdown flow to the No. 1 holding pond if the activity is above normal.

(b) Process-Waste Drain Monitor. – The flow of process-waste water is from the reactor building to monitor station No. 2. This monitor station is identical to the one described in the previous section; flow, activity, and alarm information are transmitted to the control room. Leaving the monitor station, the flow normally passes through two sets of diversion valves to the No. 1 holding pond. In the event that the activity level exceeds a preset intermediate level, the monitor station actuates diversion valves to send the flow to holding pond No. 2, which is normally kept empty. If the activity level continues to increase, a second high-level alarm will bring the situation to the operator's attention, and the flow may be diverted to the ILW holding tank. This diversion is not made automatically since the particular conditions existing at the time might not make the diversion advisable; however, the diversion may be made remotely from the control room.

(c) Intermediate-Level-Waste Monitoring. – The ILW system discharges directly to the ILW holdup tank; no in-line monitoring is provided. Sample sinks and lines are provided in the reactor building for assaying the activity level before discharge to the ILW tank for wastes which are discharged under the direct control of the operator, such as demineralizer regeneration waste. Activity in the ILW tank will be measured by laboratory analysis of dip samples. The waste will normally be sent to the ORNL waste collection system; however, if the waste is definitely of low activity, it may be sent through the process-waste system, via monitor station No. 2, to holding pond 1 or 2.

(d) ORNL Waste Monitoring Control Center. – As described in Sec. 8.7.4(c), the ORNL Operations Division maintains a central waste monitoring and control center. Information from the LWMS is transmitted to this center in order that the waste-disposal operation can be handled in an orderly and consistent manner. Personnel from the Waste Monitoring Control Center not only keep track of the flows and activity level in the various parts of the system, but they also have the responsibility for making transfers from the ILW system and the holding ponds.

8.7.5 Coolant Activity Monitoring

The CAMS is intended to provide information regarding the radioactivity present in the several coolant water systems. This information is displayed in the control room, and abnormal conditions are annunciated there. The design of this system emphasizes simplicity and reliability rather than sensitivity. Routine analysis of samples in the analytical laboratory will provide better information on low-level contamination than any presently available monitor system.

(a) Primary Coolant Monitoring. – The primary coolant monitoring system has five monitors which provide a general picture of the radioactive material distributed in the primary coolant. The design of this system presumes that an abnormal activity level in the primary coolant will most likely be the result of a fission product leak in the fuel or target region of the reactor core. An abnormal activity level could also result from leakage of an in-pile capsule or introduction of some contaminant into the coolant. Monitors are provided to detect general activity in the water stream, fission product activity in the water, and gaseous activity in the hot off-gas lines from the deaerator.

The coolant receives almost complete mixing as it passes through the core support structure and leaves via the exit pipeline. At its point of entry into the pipe tunnel three ionization chambers (RE 255-1, -2, -3) monitor the gross activity in the Faulty Fuel Element Detector system. This unit is not very sensitive to small changes in fission product or contaminant activity due to the high level of the N^{16}; however, it has been shown in several melt down type accidents that this type of activity measurement is adequate to indicate a gross activity release and is the earliest indication from activity monitors available to the operator. These three channels will trip when there is a significant increase in activity above the N^{16} background encountered at full power operation. A trip on two of these channels will produce a scram through the coincident safety system as well as supply a signal to close the letdown block valves to confine the radioactivity to the primary system.

The Cladding Failure Detector system is provided to indicate small cladding failures. A sample line is connected to the exit pipeline at a point which is only about 2 sec delayed from the core exit at normal flow. This line provides a sample stream for two separate detectors and is so arranged that the time delay from the sample point to the detectors is essentially constant despite pressure or flow variations in the primary coolant loop. The sample then enters a holdup volume where it is monitored for delayed neutrons by a BF_3 counter (RE 200) in order to provide a sensitive system for detecting fission products in the primary system. The background, and hence the sensitivity of the system, depends upon the fission products present in the primary system. The water sample is again delayed to further reduce N^{16} concentration and a GM counter (RE 253) monitors the water at this point for gross primary system activity.

In addition to the three monitors on the main coolant stream, three others are provided. The first (RE-252) monitors the off-gas from the primary coolant deaerator. This detector is sensitive to beta and gamma radiation and is provided primarily as a backup to the fission product monitor. In addition, this unit provides coverage for activity which might result from a gas leak in an experiment or from introduction of air into the primary coolant makeup water. The second monitor (RE 213) is a beta-gamma detector on the outlet of the primary coolant demineralizer units. This monitor is provided in order to detect possible malfunction in the demineralizer units by providing information on the decontamination factor of the demineralizers.

Sample points are provided at the inlet and outlet of the filters and demineralizers, and lines are brought to a sample sink where samples may be obtained for laboratory analyses.

(b) Pool Water Monitoring. – Four activity monitors are provided for the pool water systems. The four units are sensitive to beta and gamma radiation; three of them are identical to the monitor provided on the exit of the primary coolant demineralizers, and the fourth (deaerator monitor) is identical to the monitor installed on the off-gas line of the primary coolant deaerator. Of the three water monitors, one monitors the general activity of the pool coolant loop; one monitors the exit line from the defective fuel element storage tanks; and one monitors the pool demineralizer effluent. The fourth unit (RE-497) monitors the off-gas from the pool coolant deaerator.

In addition to the activity monitors, sample lines are brought from the inlet and outlet of the filters and demineralizers to a sample sink where samples may be easily obtained for laboratory analyses.

(c) Secondary Coolant Monitoring. – A beta-gamma monitor is provided for monitoring activity in the secondary coolant. This monitor is located on the inlet distribution header for the cooling tower and is provided primarily to monitor for leaks in the heat exchangers of the primary- and pool-coolant loops. This information is displayed in the control room, and an indication of activity above normal is annunciated. In addition to this, a gamma monitor is provided on the cooling tower blowdown line, as described in Sec. 8.7.4(*a*). The blowdown line comes from the same distribution header, and therefore the blowdown monitor provides backup information for the secondary coolant monitor.

8.8 Control and Safety System Analysis

8.8.1 Introduction

In order to establish the design basis for the control and safety systems discussed in the foregoing sections, the kinetic behavior of the reactor was investigated by analog simulation. These studies involved a reactor model which incorporated in essential detail the fuel region, flux trap, target, and reflector, and the coolant associated with each of these components. The effect of the hot-spot heat flux and temperature distribution; the primary coolant circuit, its pump characteristics, both main and auxiliary; the heat exchangers; the secondary coolant circuit and its pumps; and, finally, the cooling tower were all simulated.

Information generated by the analog computer included the average heat flux and the average temperature profile of the coolant in each of the regions. It was found that when at high power, the HFIR is strongly stabilized by thermal feedback in reactivity. This is due in part to the large net negative temperature coefficient of the system and in part to the extremely short time constants associated with the transfer of heat between fuel and coolant. The average fuel-to-coolant time constant at full flow is $\sim$25 msec, which is somewhat shorter than that of many pressurized-water reactors. This tight thermal coupling between the fuel and moderator together with the fuel plate prompt negative temperature coefficient provide the prompt negative reactivity feedback which is effective in quenching power excursions; in particular, it is effective in mitigating the consequences of a startup accident. In this connection it should be noted that, although the temperature and void coefficients of the coolant in the flux trap are positive, this is a region of low-power density relative to the fuel; thus, the net overall effect of increased power on reactivity is a stabilizing one at any power level.

Some of the results described below demonstrate the adequacy of the safety system under various abnormal conditions. They also make evident the ability of the control system to safely handle the reactor during certain types of coolant flow variations. The analog simulation includes most of the known mechanisms which provide internal reactivity feedback and is believed to be conservative, since it is virtually certain that additional negative reactivity factors are operative. Additional information is contained in ORNL-3573.

8.8.2 Performance Criteria

The criteria and material following set forth the information available in March 1965. Some changes in detail were made in later work; however, the same general criteria apply. ORNL-3573 sets forth information on the later work.

In order to judge the adequacy of the safety system, it was necessary to establish certain performance criteria. These criteria were based not only upon certain physical boundary conditions, but also to some extent on the experience and judgment of the designers particularly in regard to instrument response and noise. The significant criteria established are as follows.

(a) Normal Operations. – For all normal operations, including such occurrences as power failure, equipment malfunction, etc., the hot-spot heat flux is to be limited to a value which does not exceed that at which the onset of nucleate boiling occurs.

(b) Fast Excursions. – For the case of a fast excursion, two criteria have been used to judge the effectiveness of the safety system.

"Nondamage" Criterion. – The hot-spot heat flux is not to exceed the steady-state value which corresponds to the minimum burnout or incipient boiling heat flux calculated for the conditions present at the hot spot. Should this condition be reached, it is probable that some distortion of the fuel plates may occur; but no release of fission products to the primary coolant is expected to result.

"Damage" Criterion. – The fuel cladding at the hot spot reaches the melting temperature and contains just sufficient heat to actually initiate melting. In simulating this situation, it has been assumed that when the transient hot-spot heat flux exceeds the steady-state burnout or incipient boiling heat flux, the heat transfer at that spot vanishes and remains zero until the excursion is

terminated. Under these conditions, it is further assumed that the damage will be limited to melting in small areas accompanied by a minor release of fission products to the primary coolant but that no mechanical damage other than warping and distortion of the fuel plates will occur. This criterion is consistent with the observed results in the SPERT test series for cases where only a small amount of melting took place.

8.8.3 Safety System Speed of Response

In the past, reactor safety systems have often been designed primarily for protection against the startup accident (continuous motor-driven rod withdrawal from the shutdown condition). Analyses indicate that a fast-acting external safety system for this purpose is not required in the HFIR. On the other hand, a more conceivable accident would involve the introduction of a significant reactivity increase while the reactor is at power. Perhaps the most obvious way in which this could be brought about is the introduction of a void into the coolant in the target region. Whereas the geometry of the target region and its coolant is such that it is highly improbable for such a void to be swept in, the amount of reactivity and its possible rate of addition appear to represent the maximum conceivable reactivity accident to the reactor. For this reason, the "optimum void" incident has been selected as a basis for investigating the performance of the safety system.

Preliminary investigations[9] made early in the design phase indicated that the temperature coefficients alone were sufficient to handle the initial peak of all excursions resulting from a stepwise insertion at full power of up to 0.0075 $\Delta k/k$. It was found that to handle significantly greater amounts of reactivity, it would be necessary for the safety system to have a short release time and an initial acceleration in excess of $1 \times g$. As a result of these studies, it was decided that the goal of the safety system design effort should be the development of the fastest safety system considered to be practical with known techniques. The ball-latch release mechanism (see Sec. 8.2.2) originally developed for the Oak Ridge Research Reactor appeared to be the only developed release mechanism which was suitable for a drive entering from below the reactor and which also offered the desired fast-release characteristics. Based upon previous experience with this type of release mechanism, it appeared likely that a release time of 0.01 sec or less and an initial acceleration of the control plate of $4 \times g$ or greater could be achieved. Accordingly, these performance goals were established.

Investigations[10] of the several types of safety signals, trip points, etc. were undertaken in order to determine the type of signals which were most appropriate and which would offer the best protection. It was desired that the fast-acting safety system trip at the lowest power level consistent with continuity of operation. Based upon experience with fast-acting safety systems, a level trip at 1.3 times maximum allowable power (Fig. 8.1.1) was selected. In order to gain an additional margin of safety and to provide protection against the insertion of large increments of reactivity at both high and low power levels, the performance of rate trips was studied and was found to be quite effective. The system was therefore designed to trip on a rate of neutron-flux increase corresponding to a rate of power increase of 20 Mw/sec. At full power (100 Mw) this is equivalent to a 5-sec period and, at 20 Mw, to a 1-sec period. To minimize nuisance shutdowns due to electrical noise, the rate trips are RC networks rather than a differentiating amplifier; the RC time constant selected (0.25 sec) provides a noise threshold which is equivalent to a 5-Mw step in power. Large and rapid reactivity increases will, therefore, not cause a rate trip until the power has increased by ~5 Mw.

Later simulator investigations* utilized better information on reactor characteristics and were performed in order to test the ability of the safety system selected to cope with incidents occurring at full power. In these studies various amounts of reactivity were introduced on a 30-msec ramp.

*ORNL-TM-1747 and ORNL-3573.

[9] R. S. Stone, internal memorandum, August 1960.

[10] N. Hilvety, R. D. Cheverton, and O. W. Burke, *Preliminary Analysis of HFIR Transients Resulting from Ramp Reactivity Additions*, ORNL-CF-63-5-45 (May 9, 1963).

(The approximate coolant transit time in the target region is 40 msec; however, the time was reduced to better approximate the reactivity buildup.) The damage and nondamage criteria were used to assess the results. Investigations were also made to determine sensitivity to the safety-plate-release delay time, to the initial plate acceleration, and to the position of the safety plates at the initiation of a scram. The case in which one plate fails to scram was also considered. As previously stated, since the rate trip is most effective in a fast excursion, only this was used, and it was set to trip at 20 Mw/sec. The rate trip is activated at about 1.05 times normal full power if the excursion starts at full power and is very fast.

At the beginning of an operating cycle, the safety plates have a group reactivity worth of about 0.007 $\Delta k/k$ per inch. As can be seen from Fig. 8.8.1, the case of a 10-msec release time and a $4 \times g$ initial acceleration satisfies the nondamage criterion for the specified reactivity increase rate up to 0.0071 $\Delta k/k$ inserted; the damage criterion is met for reactivity increases up to 0.0142 $\Delta k/k$. It was determined that for this case, the temperature coefficients alone can handle the excursion peaks from reactivity insertions of 0.0058 and 0.0115 $\Delta k/k$ within the nondamage and damage criteria respectively.

As the operating cycle proceeds and the control plates are withdrawn, their worth per unit travel will decrease. The situation corresponding to about one and one-half days before the end of cycle is shown in Fig. 8.8.2. When these curves are compared with those of Fig. 8.8.1, it should be realized that the characteristics of the core have changed during the cycle. The prompt-neutron lifetime will have increased from about 35 μsec to about 70 μsec, and an oxide film will have built up on the fuel plates, particularly at the hot spot. For design conditions in this case, the nondamage and damage criteria yield allowable rapid reactivity increases of 0.009 and 0.0136 $\Delta k/k$ respectively.

The worst condition studied, minimum safety plate differential worth, occurs at the end of the cycle when the plates are fully withdrawn (Fig. 8.8.3). It was assumed that the withdraw limits were set so that the worth of the four shim-safety plates would be no less than 0.0007 $\Delta k/k$ per inch. Moreover, in this case it has been assumed that one of the plates fails to scram. The non-damage and damage criteria yielded acceptable reactivity changes of 0.0072 and 0.0107 $\Delta k/k$ respectively.

Another series of investigations was made to determine what, if any, requirements should be placed on the safety system in order to cope with accidents which might occur at low reactor power. The accidents studied included the startup accident and various excursions resulting from step increases in reactivity at low power. The initial power level used in these studies was $\sim$0.05 w, which is the estimated minimum power at source level. The simulation was performed both at 100 and at 10% coolant flow with the level safety system only, with the rate safety system only, and with all safety action delayed (i.e., with no fast-acting safety system).

At 100% coolant flow it was found that the reactor could satisfy the nondamage criterion during a continuous rod withdrawal of 0.005 $\Delta k/k$ per second or a step increase of 0.0098 $\Delta k/k$ even with delayed safety action. The close coupling between the fuel and the coolant and the net negative temperature coefficient provide the initial self-shutdown mechanism.

At 10% coolant flow the reduced heat transfer coefficient at the fuel plate surfaces makes the self-shutdown mechanism somewhat less effective. Moreover, the permissible hot-spot heat flux is considerably lowered by the reduced flow. In this case, the maximum startup accident which meets the nondamage criterion with delayed safety action is 0.0025 $\Delta k/k$ per second. Here the safety system is helpful, since an insertion rate of $\sim$0.005 $\Delta k/k$ per second can be handled with ease with either a 13-Mw trip level or a 20 Mw/sec rate trip.

With delayed safety action and 10% flow, a reactivity step of 0.0076 $\Delta k/k$ can be sustained without exceeding the nondamage criterion. If the safety system is operating as specified above, this is increased to 0.009 $\Delta k/k$.

It can be concluded from the foregoing that for any reasonable rate of control plate withdrawal, the safety system need not be a significant factor in terminating the initial transient induced by a startup accident.

A discussion and summary of calculational results, including comparisons with SPERT experiments, is contained in ORNL-3573.

8.8.4 Regulating System Response

As in the case of the safety system, the control system was analyzed by means of analog computer simulation; the results have been utilized to establish the performance specifications of this system.

Because of the rapid post-shutdown growth of fission product poisons, a reduction in power to compensate for an abnormal condition is highly preferable to a scram.

An analysis was made to determine the conditions under which the regulating system could reduce the power level fast enough to prevent actuation of the safety system in the event of loss of ac power to the primary coolant pumps. Upon loss of power to these pumps, the primary coolant flow will fall to 25% of its normal value in 5 sec and to ~15% in 10 sec after normal-power failure. The flow can be held at this level by the dc pony motors (see Sec. 6.5) for at least 2 hr following complete ac power failure and for an indefinite period provided the diesel generators start properly. In order to avoid a scram by the Φ_r/F circuit, the reactor power must be reduced at about the same rate as the flow. To accomplish this, the insertion of approximately $-0.02\ \Delta k/k$ is required at a maximum rate of about $-0.0015\ \Delta k/k$ per second. The regulating rod stroke is limited to $\sim \pm 0.005$ $\Delta k/k$; therefore, additional negative reactivity is needed. This is made available by the air-driven fast shim-plate insertion described in Sec. 8.3.2(*a*).

Following the power reduction described above, xenon growth will cause poisoning at a rate of about $0.002\ \Delta k/k$ per minute. While the regulating rod is utilized, it will be necessary to start withdrawal of the shim plates within 45 sec after the insertion in order to balance this growth if criticality is to be maintained. Under normal conditions ac power from the normal-emergency power systems will be available by this time; if this power is not available, the reactor will become subcritical.

8.8.5 Loss of Primary Coolant Pressure

Further analog analyses[11] were made in order to investigate the behavior of the control and safety systems following a loss of pressure to the primary coolant system. Such a pressure loss could result from a leak in the primary coolant system or from a failure of the pressure control system (see Secs. 6.2 and 8.6.4). It is clear that loss of pressure could cause burnout of the fuel at some point if the reactor continues to operate at high power. The investigation of this situation is not yet complete; however, the results to date are discussed below.

If all three letdown-flow control valves fail fully open, the letdown block valves, which are controlled by an independent pressure-sensing system, will close when the pressure drops below 550 psi. Provided that one of the main pressurizer pumps is operating, the pressure will then start to increase, causing the block valves to open again. Thus the pressure will fluctuate around a value of 550 psi. The minimum safe full-power operating pressure is much lower than the expected minimum of the pressure variations, and no scram will be necessary under these conditions.

If, in addition, the letdown block valves also fail open, the system pressure will fall to 295 psi, which will, of course, cause a scram if the trip point is set at a higher value. The value of the trip point is 375 psi, determined on the basis of field tests.

If the letdown-flow control valves fail in the closed position, the analysis shows that the system pressure increases until the pressurizer pump is no longer able to deliver any flow. This will occur at about 710 psi for normal pump speed, and neither results in a hazardous condition nor requires a reactor shutdown.

The effects on the system of various-sized leaks were also investigated. These were judged using the nondamage criterion. It was found that the time when burnout damage will occur is relatively insensitive to the location of the leak for the cases investigated. The sizes of the

[11] O. W. Burke to L. C. Oakes, "Analog Computer Analysis of Loss of Pressure Accidents in HFIR," unpublished.

simulated ruptures were specified in terms of an equivalent discharge coefficient, C_v, defined by the equation $Q = C_v\sqrt{\Delta P}$, where Q is the flow through the opening in gallons per minute and ΔP is the pressure drop across the break in pounds per square inch.

With a C_v of 5, corresponding to a rupture in one of the $\frac{3}{8}$-in.-ID instrumentation leads, the system will restabilize at about 600 psi (after a slight pressure drop), provided the pressure control system is operating. Such a leak will, however, empty the pressurizer pump suction tank in 37 min. The consequences of other leaks corresponding to valve openings of various sizes are given in Table 8.8.1.

With the letdown block valves closing below 550 psi and the pressure trip point set at 300 psi, the analysis showed that burnout can be avoided by proper choice of the pressure at which ac power is shut off to the primary pumps. This anomalous behavior (as indicated in Fig. 7.5.1) results from the fact that the incipient boiling and burnout heat flux are reduced because continued operation of the primary coolant pumps during a major low-pressure incident causes a pressure reduction at the reactor outlet. Tentative values of 375 psig for the pressure scram trip setting and 325 psig for the pump cutoff were finalized as additional analyses based on coolant system tests verified that the trip settings provide adequate protection if the pressure set point is maintained at 650 psig.

Table 8.8.1. Consequences of Primary Coolant System Leaks

Equivalent Valve Size of Leak (in.)	C_v, Equivalent Discharge Coefficient	Stable Pressure (psi)	Time Required to Empty Pressurizer Pump Reservoir (min)
$\frac{3}{8}$	5	600	37
1	10	570	14
$1\frac{1}{2}$	25	320	6
2	50	120	4.5
3	100	35	4

9. HFIR SHIELDING

9.1 General Criteria

The HFIR shielding is designed to satisfy the permissible radiation dose limitations to individuals set forth in National Bureau of Standards Handbooks No. 59 and No. 63. In unlimited access areas the shield design is such as to result in a maximum dose rate of 0.25 mrem/hr. Based upon a 50-week year and a 40-hour week, this represents just one-tenth of the annual permissible dose of 5 rem recommended by the NBS Handbooks for radiation workers. Because sustained exposure is unlikely and can be controlled, higher dose rates are permitted in limited access areas – thus making possible some economy in design.

Adequate shielding is provided not only for the reactor itself, but also for the primary coolant loop and for both the primary and pool coolant cleanup systems. Shielding is also provided, where necessary, for the various components of the hot off-gas (HOG) and special building hot exhaust (SBHE) systems. The primary heat exchangers are located in individually shielded cells and the main piping in a shielded pipe chase. The demineralizers, filters, deaerators, and other equipment are also located in shielded cells or cubicles. Shielding provided by the cells is supplemented in some cases with direct lead shielding on the equipment. The off-gas lines are shielded at all exposed points and the ventilation system filters are located in a shielded pit.

The cooling system and hot off-gas shielding have been designed not only to satisfy the requirements of normal operation, but also to handle the consequences of a major core meltdown and to permit orderly cleanup of the system without serious interruption to operations.

The maximum design dose rates for normal operation in various regions of interest are listed in Table 9.1.1. These dose rates are, in general, averages for the regions listed, and may be exceeded in small areas near the shield surface. The number and extent of these "hot spots" is kept to a minimum so that the overall dose rate is not significantly affected. It is expected that in no case will the hot spot dose rate exceed ten times that prescribed for the general area of a region. Should significant local areas of high radiation be detected, they will be 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. The general configuration is shown in Figs. 9.2.1 and 9.2.2.

9.2.1 Reactor Top Shield

The reactor core is shielded above by water, the surface of which is 27 ft above the top level of the fuel. By direct use of experimental data[1] from the Bulk Shielding Reactor (BSR), shown in

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

Table 9.1.1. Shielding Criteria

Dose rates are for normal 100 Mw operation and are for the general area

Region	Dose Rate (mrem/hr)
Beam Room Shield Face	
With beam hole plugs in place	2.5
With beam hole plugs out, beam hole flooded, shutter closed	10
Experiment Room Beam Face	
With experiment facility plugs in place and removable shielding	0.25
With experiment facility plugs out and experiment installed	<1
Reactor Bay	
Operating floor – general	0.25
Adjacent to reactor pool	1
Directly over reactor pool	2.5
Directly over reactor pool during refueling	25
Over fuel storage area	2.5
Subpile Room	10
Water Treatment Wing	
General outside equipment cells	0.25
Equipment removal corridor	1

Fig. 9.2.3, the gamma dose rate at the pool surface during operation at full power is found to be approximately 2 mrem/hr. The neutron contribution is negligible. This result does not include the additional attenuation due to the 14-in.-thick steel pressure vessel head and is therefore conservative. The thermal neutron flux at the pressure vessel wall is too low to produce a significant contribution from capture gamma rays, and the pressure vessel dimensions are large enough so that N^{16} generation in the pool is negligible.

9.2.2 Lateral Shield

Except for the region near the beam tube penetrations – which is discussed in a separate section – the lateral shield consists of 12 ft of ordinary concrete, 7 1/3 ft of water, and 1 ft of beryllium. It was estimated in the original design calculations[2] that 10 1/2 ft of concrete (plus water and beryllium) would reduce the dose rate at the shield surface to 0.17 mrem/hr even with an allowance of 5% for error in the gamma attenuation coefficients. The concrete thickness was, however, increased to 12 ft to compensate for the effect of numerous conduits and pipelines buried in the shield.

For shields of this thickness, neutrons and capture gamma rays in the shield do not significantly contribute to the dose rate at the surface. The shielding calculation is based upon gamma-ray sources originating in the reactor core. The design estimate of the dose rate at the exterior surface of the shield was made by utilizing a uniform volume source calculation applied at the

[2]Letter from C. C. Graves, Nuclear Development Corporation, to J. Russel, Singmaster and Breyer, with attached calculations.

midplane of the reactor which is the locus of the maximum dose rate. The standard equation for a uniform cylindrical volume source was employed:

$$D = \frac{BS_v R^2 f \, F(\theta, \mu t + \mu_c Z)}{2(a + Z)},$$

where D is the dose rate if:

B = symbolic buildup factor,[3]

S_v = gamma source per unit volume,

R = radius of the source region,

Z = self-shielding distance,

a = distance from edge of source region to shield exterior surface,

f = conversion factor from gamma-ray flux to biological dose rate,

$F(\theta, \mu t + \mu_c Z)$ = F function,

θ = half angle subtended by source region,

$\mu_c Z$ = self-shielding mean free paths,

μt = shield material mean free paths.

The gamma-ray sources, including capture gamma rays originating in the core, were extracted from a gamma heating calculation using the NIGHTMARE Code.[4] The total gamma power was formulated in terms of six averaged energy groups as shown in Table 9.2.1.

Table 9.2.1. Gamma Sources for Shielding Calculations

Photon Energy (Mev)	Prompt Fission and Capture	Fission Product	Total
	Mev/sec × 10^{-18}		
0.5	8.04	2.0	10.0
1.0	9.60	14.0	23.6
2.0	16.4	6.4	22.8
4.0	7.60	0	7.60
6.0	6.25	0	6.25
8.0	1.22	0	1.22

Mass attenuation coefficients of the various materials involved in the shielding calculation for each of the six energy groups were obtained from basic data[5] and are reproduced in Table 9.2.2. Self-shielding distances were obtained using attenuation coefficients calculated on the basis of a homogenized core and, because of the thickness of the concrete outer portion of the shield, the buildup factor used was that for concrete.

[3]Symbol is as defined by T. Rockwell, *Reactor Shielding Design Manual*, p. 8, Van Nostrand, New York, 1956.

[4]M. L. Tobias, *et al.*, *NIGHTMARE – an IBM 7090 Code for the Calculation of Gamma Heating in Cylindrical Geometry*, ORNL-3198 (Feb. 9, 1962).

[5]Herbert Goldstein, *The Attenuation of Gamma Rays and Neutrons in Reactor Shields*, NDA-34 (May 1, 1957).

Because the angle subtended by the reactor at the exterior surface is only 6°, the buildup factor may be introduced as a constant. Moreover, for the same reason, and because the self-shielding represents only a small fraction of the shielding, a point source calculation gives adequate results. It is possible to estimate the extreme values of the dose rate by using point sources located at the reactor axis and at the outer edge of the core region respectively.

The calculated dose rates at the reactor midplane at the surface of a concrete shield 10 ½ ft thick are shown in Table 9.2.3, both for the distributed volume source and for the extremes obtained from the point source calculations. These dose rates were computed using the attenuation coefficients given in Table 9.2.2 and would increase by a factor of about 4 if each of the coefficients were reduced by 5%.

Table 9.2.2. Mass Attenuation Coefficients (cm^2/gm)

Photon Energy	Al	Be	Fe	H_2O	Concrete[a]
0.5	0.0840	0.0772	0.0828	0.0967	0.0872
1.0	0.0613	0.0564	0.0595	0.0706	0.0637
2.0	0.0432	0.0393	0.0424	0.0493	0.0447
4.0	0.0310	0.0265	0.0330	0.0337	0.0318
6.0	0.0264	0.0211	0.0304	0.0275	0.0268
8.0	0.0241	0.0180	0.0295	0.0240	0.0242

[a] These are values for ordinary concrete with a density of 2.35 g/cc and the following composition in wt %: 0.56% H, 49.83% O, 31.58% Si, 4.56% Al, 8.26% Ca, 1.22% Fe, 0.24% Mg, 1.71% Na, 1.92% K, 0.12% S.

Table 9.2.3. Dose Rates at Lateral Reactor Shield Surface Midplane (mrem/hr)

Energy (Mev)		Volume Source	Point Source	
		Uniform	Minimum	Maximum
6		0.023	0.0053	0.056
8		0.021	0.013	0.035
	Total	0.044	0.018	0.091

9.2.3 Bottom Shield

The unpenetrated portion of the bottom shield, which provides protection for the subpile room, consists of 7 ½ ft of water beneath the reactor core, and 7 ft of barytes concrete (density 210 lb/ft^3). The dose rate through this portion of the shield has been estimated, by methods similar to those used in Sect. 9.2.2, to be 0.2 mrem/hr.

The pressure vessel extension, which is about 30 in. ID, penetrates the bottom shield on a line coaxial with the reactor core. This extension is provided with a shielding plug which begins 81 in. underneath the core and which consists of 59 in. of water contained between 18 in. of stainless steel at the top and 6 in. of stainless steel at the bottom. In addition, the bottom head provides $6\frac{1}{2}$ in. of stainless steel shielding. The maximum dose rate through the plug itself is estimated to be less than 0.1 mrem/hr during 100 Mw operation.

The bottom shielding plug is penetrated by a number of sleeves to allow access to the core for the control plate drives, the fission chambers, and the N^{16} probes. Except for the 0.063-in. water-filled annular clearance gaps, the control plate drive sleeves are plugged by the stainless steel control drives, which serve as an effective neutron and gamma shield. The other penetrations, however, may contribute significantly to the dose rate in the subpile room.

Four $\frac{3}{4}$-in.-diam pipes, surrounding the N^{16} probes, extend through the bottom plug to within 10 in. of the bottom of the reactor core. These pipes are filled with water, which is taken directly from the core exit plenum and removed through holes drilled radially in the edge of the bottom head. This head, shown in Fig. 5.4.3, provides a minimum of $3\frac{1}{2}$ in. of steel shielding for the N^{16}-bearing water. The contribution of the reactor to the dose rate beneath these sleeves was obtained by using Bulk Shielding Reactor data to obtain the dose rate at the top of the plug, and then replacing that by a Fermi-emitter with twice the strength. The local N^{16} contribution was obtained using line geometry and an N^{16} concentration of 4.3×10^6 dis sec^{-1} cm^{-3}. The hot-spot dose rate was found to be 25 mrem/hr. The N^{16} sample lines, which are $\frac{3}{8}$-in.-diam tubes, located in the subpile room are shielded with 2 in. of lead which reduces the dose rate to 16 mrem/hr at the shield surface.

The dose rate due to the three $\frac{3}{4}$-in. fission chamber tubes, which also extend to within 10 in. of the core, is extremely difficult to estimate both because the empty tubes are offset to reduce streaming, and because of the fact that the location of the fission chambers varies during operation. Based on a combination of experience and very crude analysis, a hot-spot dose rate of 100 mrem/hr seems probable. Provision has been made for local shielding against both neutrons and gamma rays should this be necessary.

9.3 Experimental Facilities

The beam tubes and the engineering facilities penetrate the main reactor shield. Special shielding is required at these penetrations to prevent leakage around and through the penetrations and to provide shielding for experiment installations which must themselves be shielded. The critical facility located at the east end of the clean pool also required separate shielding.

9.3.1 Engineering Facilities

The four engineering facilities shown in Fig. 9.2.1 penetrate the shield at an angle of about 41° with the vertical along a line tangent to the outer edge of the beryllium reflector. The outer ends of these tubes are located at the upper part of the reactor pool wall in the floor of the experiment room.

The shielding requirements and arrangement will vary depending upon the nature of the experiments to be conducted in these facilities. Other than a concrete plug of approximately the thickness of the pool wall (and the possible flooding in the rest of the tube), no permanent shielding is initially provided. Sufficient space with adequate permissible floor loading is available to permit the erection of sufficient temporary or semi-permanent shielding to reduce the dose rate from any credible experiment to less than 1 mrem/hr. Initial operation commenced with only one engineering facility; the other flanged reactor vessel openings were blanked off.

9.3.2 Beam Holes

The three 4-in.-diam horizontal beam tubes penetrate the shield at the level of the reactor center line. Two of these are tangential and one radial. The shielding for each of the beam tube penetrations is the same; however, the calculations have been based on radiation at the radial beam hole which is directed at the axis of the reactor and thus is the maximum case.

Each beam tube terminates in a recess in the beam port cavity as described in Sect. 5.6.1 and shown in Fig. 9.3.1. The portion of the shield through which the beam tube penetrates is an 8 × 7 ft section of barytes concrete. The recess penetrates into this section for about 3 ft, and the beam hole liner pierces the remaining 4 ft into the reactor pool. The liner is fitted with a stepped stainless steel sleeve which, with the water, is equivalent to about 6 ft of barytes concrete. The liner is 18 in. in diameter at the outer end and is stepped to 14 in. at its inner end. The inner bore of the sleeve is also stepped to match the aluminum beam tube. The beam tube was initially provided with a stepped plug designed to provide adequate shielding. Additional shielding may be provided by a lead barrel shutter located between the end of the beam tube and the external movable concrete shield. The shutter can provide an additional minimum equivalent thickness of 4 ft of barytes concrete when closed.

Movable outer concrete shielding which fits into the beam port cavities is provided for shielding during beam hole use. Tracks are provided on which to mount these shields. Initial operation was with stacked concrete block shielding, rather than with the movable shields.

Access to the beam tube face during reactor operation will require that the shutter be closed and that the beam tube be flooded at least a minimum distance of 8 ft. This will provide minimum shielding equivalent to 4 ft of barytes concrete plus the 8 ft of water. Using Bulk Shielding Reactor data with suitable geometry corrections, the fast neutron dose rate is estimated at 23 mrem/hr. By using a point source model, the gamma dose rate was estimated at 3 mrem/hr.

There is the possibility that some hot spots exist due to scattering through clearances around the shutter housing or through less dense aluminum portions of the shutter; however, these should not exceed 100 mrem/hr and, if necessary, local shielding can be employed.

9.3.3 Critical Facility

The design basis for the critical facility shielding was an HFIR core operated at 1 Mw. The critical facility pool, shown in Fig. 9.4.2, is 8 ft in diameter and 25 ft deep. The depth of the water over the core is 19 ft.

Direct use of the BSR data indicates a negligible contribution due to neutrons at the surface and a maximum dose rate of 17 mrem/hr due to gamma radiation exclusive of N^{16}. This rate is permissible because of the relative inaccessibility of the pool surface over the core, the ease of controlling access, and the discontinuous nature of the operation.

The polygonal lateral shield consists of the storage pool water on one side and a minimum of 6 ft of barytes concrete on the others; additional shielding is furnished by the approximate 2 ft of water between the core and the pool wall. Because of the increase in effective shielding as the angle of view through the shield increases, ordinary concrete was used in that portion of the shield above the 834-ft elevation.

Using the minimum thickness in a point source calculation, the hot-spot dose rate was found to be 2.5 mrem/hr. This occurred in an inaccessible area high on the wall of the electrical equipment room. The maximum estimated dose rate on the experiment floor was 0.25 mrem/hr located at the intersection of the shield and the floor.

Two 6-in. sleeves are provided in the shield wall to permit future installation of a pool water circulation pump, to provide for an N^{16} suppression device, or for possible experimental use. Local shielding will be required around these penetrations when in use. When not in use the sleeves will be appropriately plugged or shielded.

9.4 Coolant System Shielding

The coolant system shielding has been designed to satisfy the general criteria set forth in Sect. 9.1. In addition to these criteria, the primary water system shielding is designed to meet the following additional requirements:

1. During operation of the reactor with one defective fuel plate (pinhole leak), the maximum dose rates in unlimited access areas shall not exceed 0.25 mrem/hr.

2. A primary heat exchanger cell which is not in service may be entered on a limited time basis while the reactor is in normal operation. This implies a maximum area-wide dose rate in the cell of 100 mrem/hr, with possible higher local radiation due to residual activity in the equipment.

3. The maximum dose rate at the shield surface 24 hr after a major core meltdown shall not exceed 1 rem/hr.

4. The maximum dose rate at the shield surface immediately following a minor core meltdown shall not exceed 1 rem/hr.

For the purpose of calculation, a major meltdown is arbitrarily defined[6] as one from which the fission products listed in Table 9.4.1 are released in the amount specified to the primary coolant system. A minor meltdown is defined to be one which releases 1% of the fission products listed in this table.

The shielding thickness required for the various components of the coolant systems may be governed by any one of the criteria, depending upon which condition is controlling.

Table 9.4.1. Fission Product Fractions Released to the Primary Coolant System Following a Major Core Meltdown

Element	Fraction of Total Released	Element	Fraction of Total Released
Krypton	1	Technetium	0.01
Xenon	1	Ruthenium	0.01
Bromine	0.1	Rubidium	0.001
Iodine	0.1	Cesium	0.001
Selenium	0.01	Strontium	0.001
Molybdenum	0.01	Barium	0.001
Tellurium	0.01	All Others	0

9.4.1 Primary Coolant Loop Shielding

The thickness of the shielding for the primary coolant loop, which includes the primary pumps and heat exchangers, and the high pressure piping, is governed by the N^{16} formed in the cooling water by the $O^{16}(n,p)N^{16}$ reaction, as indicated in Fig. 9.4.1. Except for that portion of the loop located in the reactor pool, the primary loop shielding is of concrete. The general arrangement is shown in Fig. 9.4.2. The shield thicknesses are shown on the figure and are listed in Table 9.4.2.

Nitrogen-16 disintegrates by beta decay with a half-life of 7.35 sec. Approximately 75.9% of the disintegrations are accompanied by the emission of 6.13-Mev gamma rays and 6% by the emission of 7.10-Mev gamma rays. An estimate of the N^{16} concentration at various points in the system is given in Table 9.4.3. This estimate is based upon a cross section for the $O^{16}(n,p)N^{16}$ reaction of 0.59 mb for neutrons of energies greater than the effective threshold energy of 11.7

[6]G. E. Creek, W. J. Martin, and G. W. Parker, *Experiments on the Release of Fission Products from Molten Reactor Fuels*, ORNL-2616 (July 7, 1959).

Mev, upon a flow rate of 15,000 gpm, and a reactor power level of 100 Mw. The coolant circuit time external to the reactor is 25.7 sec. Design changes which resulted in higher flow rates give slightly lower N^{16} activity levels.

Other significant radionuclides which are normally present in the primary cooling water are Na^{24}, Mg^{27}, and Al^{28}, all of which result from the interaction of neutrons with aluminum. The pertinent characteristics of these isotopes are given in Table 9.4.4. The Na^{24} concentration

Table 9.4.2. Primary Coolant System Shielding Requirements

Shield Location	Items Shielded	Concrete Shielding
Wall between pipe tunnel and beam room	Piping	4 ft, barytes concrete
Wall between pipe tunnel and first pump and heat exchanger cell	Piping	$4\frac{1}{2}$ ft, ordinary concrete
Wall between pipe tunnel and rear of pump and heat exchanger cells	Piping	$4\frac{1}{2}$ ft, ordinary concrete
Wall between pipe tunnel and equipment removal corridor	Piping	$5\frac{1}{2}$ ft, ordinary concrete
Pipe tunnel roof	Piping	$5\frac{1}{3}$ ft, ordinary concrete
Walls between pump and heat exchanger cells	Piping and heat exchangers	$4\frac{1}{2}$ ft, ordinary concrete
Roofs of pump and heat exchanger cells	Piping and heat exchangers	$5\frac{1}{2}$ ft, ordinary concrete

Table 9.4.3. N^{16} Activity at Various Locations

	Decay Time (sec)	Water Activity (dis sec^{-1} ml^{-1})
Reactor vessel exit	0.0	4.3×10^6
Discharge line at tunnel entrance	1.4	3.8×10^6
Line at corner of tunnel	2.6	3.4×10^6
Inlet to Cell 110[a]	3.3	3.2×10^6
Heat exchanger in Cell 110[a]	10.3	2.1×10^6
Pump in Cell 110[a]	16.2	1.1×10^6
Inlet line at discharge of Cell 110[a]	17.5	8.2×10^5
Inlet line at corner of tunnel	21.1	5.9×10^5

[a]Activity levels in Cell 111 are 0.902 and in Cell 112 are 0.732 times those in Cell 110.

Table 9.4.4. Characteristics of Na^{24}, Mg^{27}, and Al^{28}

Isotope	Forming Reaction	Half-Life	Gammas Emitted from Isotope (Mev)
Na^{24}	$Al^{27}(n,\alpha)Na^{24}$	15.06h	1.38 (100%) 2.76 (100%)
Mg^{27}	$Al^{27}(n,p)Mg^{27}$	9.45m	0.84 (100%) 1.01 (20%)
Al^{28}	$Al^{27}(n,\gamma)Al^{28}$	2.27m	1.78 (100%)

was estimated to be between 1.8×10^8 and 7.3×10^8 atoms/ml. The latter value, which is equivalent to an activity concentration of 9.32×10^3 dis ml^{-1} sec^{-1} was used. By employing this Na^{24} activity and the relative concentrations of Na^{24}, Mg^{27}, and Al^{28} found experimentally in the ORR,[7] the concentrations of Mg^{27} and Al^{28} in the HFIR primary water system were estimated to be 3.18×10^5 and 2.01×10^5 dis ml^{-1} sec^{-1} respectively.

The shield thicknesses were calculated using the foregoing sources. In most cases, cylindrical geometry employing the formula given in Sect. 9.2.2 was used to approximate the various components in the system. A similar set of calculations with fission product sources distributed uniformly according to Table 9.4.1 was also performed. In these latter calculations the cooling system volume was taken as 11,375 gal and the core as having been subjected to 15 days operation at 100 Mw. For these calculations the four energy groups of Blomeke and Todd[8] were used to calculate the dose rates. Attenuation factors for both sets of calculations were those of Snyder and Powell.[9] Some typical results of these calculations are shown in Fig. 9.4.1.

9.4.2 Primary Cleanup System Shielding

Shielding for the primary coolant cleanup system is provided to afford protection from activation products such as Na^{24} which are removed by the system, and in addition to permit orderly decontamination of the primary coolant in the event of a core meltdown. The various components of the system are located in shielded cells, the arrangement of which is shown in Figs. 9.4.3 and 9.4.4. The shielding requirements are listed in Table 9.4.5. In some cases stacked block rather than poured concrete is used in order to facilitate repair or replacement of the various items of equipment.

Water entering the primary coolant cleanup system through the let-down lines is held up in a decay tank for at least 90 sec before it enters the primary coolant deaerator. This is sufficient delay to allow decay of N^{16} to a negligible concentration. The decay tank is a long, 12-in.-diam pipe located in the pipe tunnel and is therefore protected by the primary loop shielding.

Shielding for the deaerator is governed by the quantity of volatile fission products which may possibly accumulate in this unit following a major fuel meltdown. The shielding calculations are based on the deaerator's containing all of the Kr, Xe, I, and Br released as indicated in Table 9.4.1. The results indicate that 3 ft of barytes concrete is adequate to satisfy the meltdown criteria and

---

[7] J. C. Ward, *Radioactivity of Nuclear Reactor Cooling Fluids*, ORNL-3152 (Sept. 20, 1961).

[8] J. O. Blomeke and M. F. Todd, *Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time and Decay Time*, ORNL-2127 (Aug. 19, 1957).

[9] W. S. Snyder and J. L. Powell, *Absorption of Gamma Rays*, ORNL-421 (Mar. 14, 1950).

Table 9.4.5. Primary Coolant Cleanup System Shielding Requirements

Shield Location	Items Shielded	Shielding
Letdown line in equipment removal corridor	Letdown line	3 in. lead
Primary coolant deaerator cell	Primary water lines, deaerator, and deaerator condensers	3 ft barytes concrete
Primary coolant cleanup system pump cell	Primary coolant cleanup system pump	2 ft ordinary concrete
Primary coolant cleanup system cell	Primary cation exchanger	$3\frac{1}{2}$ ft ordinary concrete plus 3 in. lead
Primary coolant cleanup system cell	Primary anion exchanger	$3\frac{1}{2}$ ft ordinary concrete
Primary coolant cleanup system cell	Primary coolant system prefilter	$3\frac{1}{2}$ ft ordinary concrete plus 3 in. lead
Primary coolant cleanup system cell	Primary coolant system afterfilter	3 ft ordinary concrete

more than adequate to satisfy the criteria for normal operation. As before, cylindrical geometry was used in these calculations.

Shielding for the primary coolant cation exchange units is based on the assumption that all of the Na^{24}, Mg^{27}, and Al^{28} in the bypass flow to these units is removed. The total flow to each of the two units is taken to be 200 gpm and the isotopic concentrations are those given in Sect. 9.4.1. The shield thickness was calculated using cylindrical geometry. The shielding provided – a $3\frac{1}{2}$-ft ordinary concrete cell wall plus 3 in. of lead cladding on the cation exchanger vessel – is sufficient to reduce the dose rate at the exterior cell wall to 0.13 mrem/hr. This was found to be more than adequate to meet the meltdown criteria also.

Shielding for the anion exchange units was based upon the assumption that all of the iodine released during a major meltdown was absorbed in the two units. In this case, it was found that $3\frac{1}{2}$ ft of ordinary concrete was adequate to satisfy the meltdown criteria and more than adequate to meet the requirements of normal operation.

The primary coolant prefilters are located in the same cells with their respective ion exchange units, and they are shielded individually with 3 in. of lead cladding in addition to the $3\frac{1}{2}$-ft thick cell wall. This has been shown to be adequate for both the meltdown and normal operation cases. The afterfilter will accumulate little activity; it is located in one of the demineralizer cells and the $3\frac{1}{2}$-ft concrete cell wall provides adequate shielding.

The letdown lines which convey water from the primary loop to the primary coolant system are, where exposed, shielded by 3 in. of lead. All other lines and equipment are either located in the equipment cells or buried in the shielding walls. Calculations show that the equivalent of 2 ft of ordinary concrete is adequate for these items.

9.4.3 Pool Cleanup System

The shielding requirements for the pool cleanup demineralizers and filters were based on experience with the ORR system. The radiation levels at the surfaces of the ORR pool filter and demineralizer range from 6 to 14 mrem/hr. It was calculated that 1 ft of ordinary concrete would be sufficient to shield these units in the pool cleanup system except for the cation exchanger, which requires 2 in. of lead shielding in addition to the 1-ft concrete wall.

The shielding for the exposed portion of the discharge line from the defective fuel element storage tanks to the pool cleanup system was based on the arbitrary assumption that 0.1% of a melted fuel element is located in this portion of the line. It was found that $2\frac{1}{2}$ in. of lead would reduce the resulting radiation to 1 rem/hr 5 ft from the line; so this line is either buried in an equivalent thickness of concrete or shielded with this thickness of lead in accessible spots.

9.5 Ventilation System Shielding

The HFIR containment (SBHE) and hot off-gas systems (HOG), described in Sec. 4, are designed to remove airborne radioactivity from the HFIR building and equipment and deposit particulate activity on filters which are located in a pit just outside the northwest corner of the reactor building.

No special shielding is provided for the SBHE ducts within the building; however, much of the ductwork is buried in the various concrete walls or inside a shielded compartment. Where convenient, the HOG lines are also buried in concrete or, in exposed areas, are shielded by the equivalent of 1 in. of lead. Provision is made for additional local shielding should this be found necessary because of the requirements of some of the experiments or other equipment.

Relatively large quantities of radioactive material may be collected on the filters. The shielding of the filter pits, summarized in Table 9.5.1, was calculated by using all of the iodine and bromine released during a major meltdown as the radiation source. For the purpose of computation, it is assumed that these products are uniformly distributed as a line source 9 ft in length for a SBHE filter bank and 2 ft in length for an off-gas filter bank. The shielding is adequate to permit filter removal from a pit which is off-stream during normal reactor operation.

Table 9.5.1. Filter Pit Shielding

Shield Location	Items Shielded	Shielding
Roof of the special building hot exhaust (SBHE) filter cells	SBHE filter	2 ft, ordinary concrete
Partitions between SBHE filter cells	SBHE filter	2 ft, ordinary concrete
Sides of SBHE filter cells	SBHE filter	2 ft, ordinary concrete for a distance of at least 2 ft below ground level
Manhole covers over the inlet and outlet lines of the SBHE filter cells	Inlet and outlet lines of the SBHE filter cells	1 ft, ordinary concrete
Roof of the hot off-gas (HOG) filter cells	HOG filter	4 ft, ordinary concrete
Partitions between HOG filter cells	HOG filter	4 ft, ordinary concrete
Partitions between SBHE filter cells and HOG filter cells	HOG filter	4 ft, ordinary concrete
Sides of HOG filter cells	HOG filter	4 ft, ordinary concrete for a distance of at least 4 ft below ground level
Manhole covers over the inlet and outlet lines of the HOG filter cells	Inlet and outlet lines of the HOG filter cells	2 ft, ordinary concrete

Because activity entering the SBHE filters would result from an underwater meltdown, the shield thickness was specified on the basis of a source equivalent to 1% of the total halogens in the core, and the HOG filter shield thickness was based on 10% of the halogens.

The HOG fans may be shielded by 2 ft of stacked ordinary concrete block to provide protection against radiation from activity which may collect on the fan surface.

9.6 Spent Fuel Shielding

When spent fuel elements are removed from the reactor they must be stored, Fig. 9.6.1, for 1 day on a specially shielded rack in the reactor pool and then may be transferred to a rack in the storage pools for further fission product decay prior to shipment to a reprocessing plant. Any fuel element that has a cladding failure will be put into a defective fuel element storage facility located in the reactor pool under 30 ft of water, which will prevent contamination of the pool water by flowing water from the pool through the defective element to the pool cleanup system.

The spent core storage racks, Fig. 9.6.2, are located on $2\frac{1}{2}$-ft centers and support the spent cores 2 ft above the pool bottom. The racks position the cores a minimum of $7\frac{1}{2}$ ft from the south pool wall (5-ft-thick concrete) and a minimum of $3\frac{1}{2}$ ft from the north wall (4-ft-thick concrete). The experiment room is located south of the pool and is an unlimited access area. Behind the north wall are the heat exchanger cells which are accessible on a limited time basis only.

The defective element storage tanks have special steel shadow shields 3 in. thick, Fig. 9.6.3, to prevent overheating of concrete and streaming into the HB2 beam path.

A stairwell leading to the subpile room is located beneath part of the clean storage pool. Shielding for this stairwell is provided by the pool floor which is composed of 22 in. of barytes concrete on top of 14 in. of steel armor plate.

9.6.1 Estimated Dose Rates

Estimated hot-spot dose rates at the outer edge of the shield for handling and storage of spent cores are shown in Table 9.6.1 for the areas described. The gamma sources and pertinent attenuation parameters are given in the following section.

In these calculations, the core was considered a point source; for the distances involved, the point source approximation is a good one and errs on the conservative side.

Table 9.6.1. Estimated Dose Rates from Spent Cores

Operation	Shield	Dose Location	Dose Rate (mrem/hr)
Transfer	$14\frac{1}{2}$ ft water	Reactor pool[a]	14
Reactor pool storage	16 ft water	Reactor pool[a]	2
Clean pools storage	16 ft water	Clean pool[b]	0.1
Clean pools storage	5 ft concrete plus $7\frac{1}{2}$ ft water	Experimental area[b]	0.05
Clean pools storage	4 ft concrete plus $3\frac{1}{2}$ ft water	Heat exchanger cell[b]	100
Clean pools storage	22 in. barytes concrete plus 14 in. steel	Stairwell to subpile room[b]	1.5

[a]One hour after reactor shutdown.

[b]One day after reactor shutdown, freshest core in worst location.

The results are for one core that was used for 15 days of 100-Mw operation and allowed to cool 1 hr or 1 day as indicated. Older cores stored in adjacent positions could add only about 25% at most to the hot-spot dose rate. The indicated 100 mrem/hr hot spot in a heat exchanger cell will present no problem, since transfer can be made to an interior rack if access is required when this particular storage arrangement occurs.

9.6.2 Gamma Sources and Attenuation Parameters

The data of Perkins and King[10] were used to estimate the gamma activity of the spent cores. Table 9.6.2 shows the gamma energy released for each energy group used in the calculations.

The mass attenuation coefficients[11] of the materials involved for these shielding calculations are shown in Table 9.6.3.

Table 9.6.2. Gamma Energy Release from Spent Cores, Following Fifteen Days Operation at 100 Mw

Mean Energy	Gamma Energy Released (Mev/sec)	
	One Hour After Shutdown	One Day After Shutdown
1.3	4.4×10^{17}	8.0×10^{16}
1.7	5.9×10^{17}	1.6×10^{17}
2.2	3.8×10^{17}	3.2×10^{16}
2.5	1.8×10^{17}	1.6×10^{16}
2.8	5.5×10^{16}	5.1×10^{14}

Table 9.6.3. Mass Attenuation Coefficients (cm^2/g)

Photon Energy (Mev)	Water	Steel	Concrete[a]
1.3	0.0621	0.0560	0.0525
1.7	0.0541	0.0484	0.0458
2.2	0.0469	0.0421	0.0420
2.5	0.0437	0.0383	0.0382
2.8	0.0410	0.0368	0.0366

[a]These values are for ordinary concrete and when used for barytes concrete give slightly conservative results.

For shields composed of more than one material, the buildup factor for the outermost material was used since in all cases the number of mean free paths of the outermost region exceeded 3. The self-shielding factors used in all cases were for cylinders for a dose exterior to the side. These values are conservative for dose rates exterior to the ends of a solid cylinder the size of the HFIR core; however, the water-filled island, which is less dense than the fuel region, tends

[10]J. F. Perkins and R. W. King, "Energy Release from Decay of Fission Products," *Nucl. Sci. Eng.* 3(6), 726–46 (June 1958).

[11]Herbert Goldstein, *The Attenuation of Gamma Rays and Neutrons in Reactor Shields*, NDA-34 (May 1, 1957).

to remove the conservativeness. This simplified procedure is justified since the self-shielding factor is of small importance.

9.7 Shielding During Maintenance

Access to the pipe tunnels and any heat exchanger cell in use will be prohibited during reactor operation. Even after shutdown, access to these areas must be limited and controlled since gamma activity due to corrosion product deposition will exist in all primary cooling water piping. The dose rate from this source cannot be predicted; experience with other reactor systems indicates that it may be as high as a 100 mrem/hr at the surface of the heat exchanger.

Access to the demineralizer and deaerator cells during operation will be possible on a very limited and controlled basis. Depending on the location of access and type of maintenance, some portable shadow shielding may be required. Design of such shielding is left to the operating organization so that it may be tailored to the particular job.

10. UTILITIES

10.1 Electrical Systems

The HFIR is supplied electricity by four semi-independent systems which are illustrated in Fig. 10.1.1. These four systems are identified as:

1. The normal-power supply system, which handles the normal electrical power requirements. This system is supplied with TVA power from one of two independent 13.8-kv feeders.
2. The two normal-emergency systems (designated No. 1 and No. 2) which are designed to assume certain electrical loads during an outage of the normal-power system. Each normal-emergency system is supplied by its own start-on-demand diesel motor-generator set.
3. The instrument-power system, which is supplied with current from the 13.8-kv feeder through shielded low-electrical-noise transformers or from normal-emergency system No. 1. The instrument-power system furnishes power for reactor and experiment instrumentation.
4. The failure-free systems, which are supplied with current (dc) from a bank of batteries. The charge on the batteries is maintained by normal-emergency system No. 1 or 2.

The TVA network supplies power (13.8 kv) to the normal-power system via the ORNL substation. From the 13.8-kv bus, the power is distributed to five substations and two transformers. Substation No. 5 furnishes 2.4-kv power to the larger motors. The remainder of the plant receives 480-v power from the other four substations, either through bus ducts or through one of the motor control centers (MCC). Further reduction in voltage is accomplished, where necessary, by transformers in the motor control centers. The two low-noise transformers, designated No. 6 and No. 7, receive current directly from the 13.8-kv bus and furnish power for the instrument-power system.

Those components and services whose continued operation during a normal-power outage is deemed desirable or necessary to plant safety are furnished electrical power through a normal-emergency system. Upon failure of normal-power, these essential components receive current from a diesel-driven generator. The components connected through the normal-emergency systems include the special building hot exhaust (SBHE), open hot off-gas (OHOG), and closed hot off-gas (CHOG) fans; the slow-speed winding of the secondary coolant pump; shim-safety plate drive motors; the failure-free system battery chargers; the auxiliary pressurizer pump; certain motor-operated valves; the instrument air compressors; and the fire alarm and communications systems. Where operation is vital, two identical components are installed with one supplied by each diesel-driven generator.

Those components whose operation during a power outage is considered of greatest importance are supplied by the failure-free system. Equipment in this category includes the primary coolant pump pony motors, automatic electrical switchgear, regulating cylinder drive motors, and certain of the reactor instruments. These components are supplied with dc from batteries which are capable of delivering adequate power for at least two hr even in the event of failure of the normal-emergency system. The batteries are kept charged by appropriate rectifiers and controls. Eight separate failure-free systems are used: one for each of the four pony motors, one for each of the three reactor instrument buses, and one for the automatic, electrical, transfer switchgear.

10.1.1 Normal-Power System

The normal-power system consists of two 13.8 kv, three-phase feeders which are run to the electrical building on separate sets of poles. The preferred feeder has a capacity of 10,000 kva.

The alternate feeder is capable of maintaining the full plant load (~7000 kva) for about two hr at any time and can handle the load indefinitely by a rearrangement of other loads normally on this line. Upon failure of the preferred feeder, the 13.8-kv bus is automatically switched to the alternate feeder. This 13.8-kv switching station consists of an indoor, completely enclosed, metal clad, assembly which includes two 13.8 kv, 1200 amp, electrically operated air circuit breakers, six 600 amp manually operated interrupter switches which serve the substations and transformers, and the requisite control equipment for the automatic transfer scheme.

Power from the 13.8-kv switchgear is supplied to seven transformers. One of the transformers (Fig. 10.1.2), designated substation No. 5, supplies 2400-v power for the primary coolant pumps, the secondary coolant pumps, the main pressurizer pumps, and the central chilled-water compressor. Four transformers, designated substations 1, 2, 3, and 4, supply 480-v power to the HFIR area, and the other two transformers, 6 and 7, supply 120/240 v instrument power. Three of the 13.8-kv–480-v transformers and the 13.8-kv–2400-v transformer are located outside the west wall of the electrical building. The other 13.8-kv–480-v transformer (substation No. 1) is located outside the north wall of the reactor building. The instrument transformers are located inside the electrical building. Connections between the 13.8-kv switchgear and the transformers are made with 3-conductor, paper-insulated, lead-covered, neoprene-jacketed cable, run in underground concrete-encased conduit. The outside transformers are mounted on concrete pads which include a curb for oil containment. Fire walls (if required) are installed between adjacent units.

The 13.8-kv–480-v voltage reduction units include an oil-filled transformer with an integral oil-filled disconnect switch, a 460-v metal-clad switchgear assembly, and a metal-enclosed bus duct between the transformer and switchgear. The transformers are rated at 13,800 v, wye-connected, 480-v delta, 1000 kva, 55°C rise, self-cooled, 3-phase, and 60 cps. Provisions have been made for the addition of fans which can increase the rating to 1250 kva. The 480-v switchgear consists of manually operated main and feeder breakers in three vertical stacks. Power distribution is 480-v delta (ungrounded) with ground detection monitoring lights. Appropriate voltmeters and ammeters are provided.

The 13.8-kv–2400-kv unit includes an oil-filled transformer, a lineup of high-voltage fused motor starters, a main transformer secondary breaker, and a feeder breaker. The transformer is rated at 13.8-kv–2.4-kv delta-delta, 5000 kva self-cooled, 3-phase, and 60 cps. Forced-air cooling increases the rating to 6667 kva. The 2400-v ungrounded delta system is converted into a resistance grounded system by a zig-zag grounding transformer and resistor.

Power from the transformers is distributed to the various loads through thirteen motor control centers (MCC) and a number of bus ducts. The distribution which shows the power sources for the principal plant loads is given in Table 10.1.1. Note that the loads from substation No. 3 which are routed through motor control centers *D, E, G,* and *J*, constitute normal-emergency system No. 1. Those loads from substation No. 4 which are routed through motor control centers *F* and *H*, together with the 250-amp experiment area bus, constitute normal-emergency system No. 2.

10.1.2 Normal-Emergency Systems

The purpose of the normal-emergency systems is to provide power for certain essential equipment in the event of a normal-power system outage. This is accomplished by two diesel-driven generators which are connected in such a way that upon failure of normal-power, one of the diesels will assume the loads in a normal-emergency system. There is no connection between the two normal-emergency systems, except for a generator test load. The principal components and systems served by the normal-emergency systems are listed in Table 10.1.1 and are shown in greater detail in Fig. 10.1.3a and b and Fig. 10.1.4.

All normal-emergency system circuits are supplied from the normal-emergency switchgear located in the electrical building. During routine operation this switchgear is energized by 480-v feeders from substations No. 3 and No. 4; the diesel generators are not running, but their main

Table 10.1.1. Normal-Power Distribution System

Routing	Principal Loads
Substation No. 1 (480 v)	
MCC *A*	Plant demineralized water pumps, reactor building central air conditioning, and reactor bay crane
MCC *B*	Primary coolant cleanup and pool demineralizer pumps, pressurizer pump couplings, pool coolant pumps, and letdown block valves
MCC *O*	Office and maintenance building
MCC 1	SBHE fan 3
Substation No. 2 (480 v)	
MCC *C*	Heat exchanger cell unit coolers
600 amp bus	Experiment room air conditioners
600 amp bus	Beam room air conditioners
225 amp bus	Experiment distribution panels
Substation No. 3 (480 v)	
MCC *N*	Street lighting, chemical treatment lighting, normal winding of auxiliary secondary coolant pump and half the cooling tower fans
Lighting Panel P-26	Reactor building lighting
Normal-Emergency System No. 1	
MCC *D*	Primary and secondary coolant valves on primary coolant heat exchangers, battery chargers for pony motors, pool bridge drive, and shim-safety plate drive motors
MCC *E*	Process water pumps, pool cooling system valves, auxiliary pressurizer pump, two instrument air compressors, truck air lock doors, fire alarm and communications systems, and emergency building lighting
MCC *G*	One each SBHE, CHOG, and OHOG fan
MCC *J*	Emergency instrument power transformers 6A and 7A, emergency windings of auxiliary secondary coolant pump, intermediate level waste valves, and diesel services
100 HP TRU Feeder	
Transformers 6A and 7A	Standby transformers for instrument power supply and relays for the regulating cylinder and shim-safety plate drive motors.
Substation No. 4 (480 v)	
MCC *K*	Half the cooling tower fans and one fire-protection system air compressor
MCC *L*	Diesel services
Normal Emergency System No. 2	
MCC *F*	Intermediate level waste pumps, diesel services, and emergency outside lighting
MCC *H*	One each SBHE, CHOG, and OHOG fans; stack monitors
225 amp bus	Emergency air compressor and instrument air compressor, and battery charger for pony motors
100 HP TRU Feeder	
Substation No. 5 (2400 v)	Four primary coolant pumps, three secondary coolant pumps, two main pressurizer pumps, and central air conditioning chiller
Transformers 6 and 7 (120/240 v)	Instrument power supply and relays for the regulating cylinder and shim-safety plate drive motors

breakers are closed. In the event of a sustained normal-power supply outage of approximately two-sec duration, the diesels are started automatically.

(a) Diesel Generators. – Diesel No. 1 is equipped with two 32-v dc starting motors. Power is supplied by a 24-cell nickel-cadmium battery. A constant-voltage battery charger connected to normal-emergency system No. 1 keeps the batteries fully charged. Diesel No. 2 is equipped with an air-motor starter. The air motor is designed for operation over a range of 90 to 150 psi. An air receiver with sufficient capacity for five starts of approximately fifteen-sec duration each is also provided. The receiver pressure (250 psi) is reduced to approximately 100 psi by a regulator on the air-motor inlet.

The four-cycle diesel engines, located in the electrical building, have a rated capacity of 625 hp. They are designed to operate on No. 2 fuel oil. The fuel oil supply system includes a 4000-gal fuel storage tank, an individual 60-gal day tank for each unit, and associated pumps and piping required for oil transfer. The 4000-gal storage tank is buried in the ground south of the electrical building. Oil is transferred from the storage tank to the day tanks by either of two 60-gph positive displacement pumps located at the respective day tanks in the electrical building. The fuel transfer pumps are powered by the respective normal-emergency system and are automatically started and stopped by level controls in the day tanks. The pumps and day tanks are interconnected with piping to provide fuel to both engines in the event of failure of a single transfer pump. Each 60-gal tank provides sufficient fuel to run one engine for about two hr.

The diesel generators are 1200-rpm, 6-pole, 3-phase, 60-cps units rated at 350 kw; that is, 437 kva at a 0.8 power factor. The generators are capable of supplying 110% of rated load for two hr without serious overloading. Each generator unit contains an integrally mounted static exciter, regulator, and control panel which provide a steady-state voltage regulation of $\pm 2\%$ of the rated voltage.

(b) Load Switching. – A diesel generator is started when the voltage in any phase of its normal-emergency system power supply drops below 70% of normal for a sustained period of approximately two sec. The load is transferred to the diesel-generator supplied circuit automatically by means of the 3-pole, 480-v, 600-amp transfer switches. Auxiliary contacts in the transfer switches start the diesels as in the following description:

1. Normal-Emergency System No. 1
   a. Energize the startup relay to crank the engine.
   b. Open all the circuit breakers which supply loads through normal-emergency system No. 1 except the TRU feeder.
   c. Open the test load circuit breaker. (This is to remove the test load in the event that the power failure occurs during a diesel engine test.)
2. Normal-Emergency System No. 2
   a. De-energize the solenoid valve; thereby permitting air to be supplied to the starting motor (two parallel valves are provided).
   b. Open all circuit breakers which supply loads through normal-emergency system No. 2 except the circuit breaker to MCC *H* which feeds the stack fans.
   c. Open the test load circuit breaker.

As soon as the diesel generator has delivered approximately 90% of rated voltage, the various load circuit breakers automatically close (in a timed sequence) to restore current to the loads. The timed closing sequence is as follows:

Normal-Emergency System No. 1

0 sec	one TRU feeder
10 sec	MCC G – one each SBHE, OHOG, and CHOG fan
20 sec	MCC *J* – instrument power transformers 7A and 6A, the auxiliary secondary coolant pump's emergency winding, and liquid waste diversion valves
	MCC *D* – primary and secondary coolant valves for the PC heat exchangers, and failure-free systems for PC pump pony motors

30 sec MCC *E* – process water pumps, pool coolant system valves, auxiliary pressurizer pump, two instrument air compressors, and fire alarm and communications systems
40 sec 225 amp bus duct for experiments

Normal-Emergency System No. 2
0 sec MCC *H* – one each SBHE, OHOG, and CHOG fan
10 sec one TRU feeder
20 sec MCC *F* – ILW pumps and diesel services
30 sec one instrument air compressor and emergency air compressor and pony motor battery charger receptacles

Upon restoration of the normal power supply for approximately 30 sec, the transfer switch automatically returns the load to the substations and the 13.8-kv normal-power supply.

The diesel generators may be tested by connecting them to the building lighting system. This loads a single generator [redacted] capacity without interrupting the normal-emergency distribution system. The test circuit is designed so that either, but not both simultaneously, of the generators can assume the lighting load. In the event of a normal-power failure during the test, the test load is automatically disconnected from the generator.

10.1.3 Instrument-Power System

Power for the reactor and experiment instruments, as illustrated in Fig. 10.1.5, is obtained from the 13.8-kv switchgear through a fused interrupter switch which supplies two 13.8-kv–120/240-v, single-phase, transformers. Transformer No. 6 supplies power for experiment instrumentation and radiation monitors. Transformer No. 7 supplies power to the shim rod drive relays and, through a failure-free system of batteries, to the reactor instruments and the regulating cylinder drive motors.

In the event of failure of normal-power or of an instrument power transformer, the instrument load is automatically transferred by means of an automatic 3-pole, 600-v, 300-amp transfer switch, to the normal-emergency No. 1 bus. The instrument load is then fed through two 480-120/240-v standby transformers – No. 6A and No. 7A. A failure of one of the instrument transformers will not start the diesel automatically, since normal-power is still available on normal-emergency system No. 1. Return to the regular instrument-power transformer, following a power outage, is automatic upon restoration of normal-power for a period of 30 sec.

The main instrument transformers are 13.8 kv–120/240-v single phase, 60 cps, and are rated at 50 kva. In each case, the transformer secondary winding is electrostatically shielded from the primary by a copper shield. The standby transformers are 480-120/240-v single phase, 60 cps, and are also rated at 50 kva. The standby transformers are of standard nonshielded construction. The transfer switch is provided with a test switch to simulate normal-power failure; thereby permitting a test of the system.

10.1.4 Failure-Free System

A power supply is provided which will remain uninterrupted for a sufficient period of time to permit orderly shutdown of the system. This power supply is necessary to prevent damage to certain critical components should all other power sources fail. This failure-free power supply is provided by battery banks, which are kept charged by current from either normal-emergency system No. 1 or the main instrument-power transformers. The energy stored in the batteries will provide sufficient power to shut down the reactor in a safe, orderly manner. Components fed by the failure-free systems are the primary coolant pump pony motors, the reactor nuclear instruments, regulating cylinder motors, and certain of the normal-emergency and 13.8-kv switchgear.

Each of the primary coolant pumps is equipped with a 3 hp, dc pony motor to provide forced convection cooling in the event of failure of the 2.4-kv power to the primary coolant pump motors (Fig. 6.2.8). Although failure of the main pump motors will cause an automatic reduction in reactor power and, in some cases, reactor shutdown, forced convection cooling is required (continuously in the case of a power reduction or for about 4 hr following a reactor shutdown) to prevent damage to the core from overheating (see Sec. 6.5).

The pony motors are energized whenever the reactor is put into operation, thus eliminating the possibility of switching failures. Power to each pony motor is supplied from a battery and battery charger in parallel. Each battery bank will provide 120-v power from a 92-cell, nickel-cadmium, alkaline-pocket, plate type, battery. The chargers, which receive current through normal-emergency system No. 1, are the constant voltage type with silicon diodes. They are designed to supply 50 amps continuously at 129 v. This is sufficient to carry simultaneously the full-load current of the pony motors and completely recharge the batteries in approximately 12 hours. Automatic controls provide a "freshening" charge during normal operation. If the No. 1 normal-emergency system fails, the chargers may be manually connected to normal-emergency system No. 2 using quick-disconnect connections already installed at the chargers.

The No. 7 instrument-power transformer (Fig. 10.1.5) supplies current to three separate pairs of battery chargers which, in turn, maintain charge on three pairs of 32-v battery banks. Each battery bank is a 24-cell, nickel-cadmium assembly similar to those used for the pony motors. The battery charger is a constant voltage type with silicon diodes rated at 50 amps continuously at 32 v dc. Each pair of batteries is connected so that it can supply either 32 v or 64 v of dc. These batteries, in turn, feed the three reactor instrument buses and the three servo-control supplies.

Upon failure of the power supply from transformer No. 7, the charger load is picked up by normal-emergency system No. 1 through standby transformer No. 7A. Upon failure of this power source, the batteries will supply instrument power for several hours.

An independent power supply system is required at the electrical building to operate the normal-emergency and 13.8-kv switchgear and diesel generator control circuits during a normal-power failure. These are operated by dc power from a battery, with a charger supplied from MCC *J*. Power is supplied from a 125-v, nickel-cadmium battery. The charger is a constant voltage type with silicon diodes and supplies 12 amps continuously at 129 v dc.

10.2 Plant Water Systems

The HFIR plant water supply, shown in Figs. 10.2.1 and 10.2.2, is obtained through either of two 16-in. water mains. The preferred water main is fed from a 3 million gal (potable) water reservoir located on Haw Ridge at an elevation of 1035 ft, and is capable of supplying 2800 gpm. Normal plant usage is 1000 gpm. This reservoir receives water directly from the Y-12 water treatment plant. The alternate water main is fed from the ORNL (potable) water reservoir located just north of the main ORNL area at an elevation of 1000 ft. This water main can also supply the normal requirements of the HFIR facility.

The potable water feeds the fire hydrants in a ring-main system. It also supplies drinking water, plumbing needs, secondary coolant system make-up, and the cooling requirements of the instrument air compressors and the pressurizer pumps variable speed drives.

Supply water to the plant demineralizer system and various experiment requirements is fed from the potable water system through an "air gap" tank. The "air gap" tank provides positive protection against back-flow from these particular uses into the potable water loop. Similar protection is provided at the cooling tower basin. In other cases where back-flow is undesirable but could not cause radioactive contamination of the potable water system, protection is provided by a back-flow preventer. 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 through one of the two 16-in. mains to the fire loop. It is then distributed (for the purpose of fire protection) to the reactor building, the cooling towers, the electrical building, and the office and maintenance building. Water to supply plumbing needs is also distributed from this loop to the various buildings. Water for secondary coolant make-up is carried in an 8-in. line to the cooling tower basin where it is discharged one ft above the maximum water level of the cooling tower basin, thus providing air-gap separation. A 2-in. branch line to the Cooling Tower Equipment Building supplies water through a back-flow preventer to the chemical mixing equipment, the safety shower, the eye-washer, and the sink. Water to cool the air compressors and primary pressurizer pump variable speed drive units is also taken directly from the potable water system.

10.2.2 Process Water System

The 6-in. potable water line entering the north side of reactor building from the fire loop has two branches, one to the domestic potable water system described above and a second to the 1000 gal air-gap tank located in room 103 on the first floor of the reactor building. Potable water enters this tank through two lines near the top of the tank. The tank is vented to the atmosphere. The liquid level is controlled by two automatic valves which maintain a minimum air-gap of one ft between the liquid surface and the inlet lines. Water which has reached this tank is termed process water. Additional overflow protection is provided by a drain line to the process waste system.

Process water is supplied to the plant demineralizers, experiments, and other facilities by two centrifugal pumps (PU-17A and B) which take their suction from the air-gap tank. These pumps, one of which is used as a standby, are rated at 500 gpm at 175 ft head. Each pump has a standby start feature – a low pump discharge pressure will automatically start the standby pump. In addition, a low-level switch in the air-gap tank will alarm in case of inadequate water level in the tank. Pump motors receive power through normal-emergency system No. 1.

The process water pumps discharge to a 6-in. header which supplies the process water system branch lines in the reactor building, the filter pits, and the intermediate level waste transfer pit. A small water storage tank is connected to that branch which supplies process water to the experiments. This tank is kept pressurized by air compressed in filling the tank and will supply a minimum of 200 gal of water at an average pressure of 43 psi during the delay time necessary to start the diesel engines.

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 demineralized water for miscellaneous plant applications.

The plant demineralizer is a package monobed unit which has a maximum flow rate of 90 gpm. The 3-ft-diam, 9-ft-high demineralizer vessel is fabricated of carbon steel and lined with an 80-mil coating of unplasticized polyvinyl chloride. It contains 13 cu ft of IR-120 cation resin and 25 cu ft of IR-402 anion resin. It is designed to demineralize approximately 45,000 gal of process water between regenerations and to produce effluent having a specific resistivity of at least 1.5×10^6 ohm-cm. The flow rate can be varied between 10 and 90 gpm with no appreciable change in effluent water quality. The unit includes control instrumentation to permit semi-automatic operation and will shut down automatically when the effluent resistivity exceeds a preset level. The shutdown is annunciated in the control room.

The demineralizer receives process water from the process water pumps and discharges demineralized water into a 20,000 gal aluminum storage tank which is located above ground just outside the northeast corner of the reactor building. A level signal from this tank is transmitted to an indicator on the process panel board in the control room. High and low water level alarms are also annunciated in the control room. A flow control valve on the 2-in. outlet line of the demineralizer is controlled by the level of the water in the storage tank. This mode of control tends to hold the tank level constant by varying the flow through the demineralizer according to the system demand.

Demineralized water from the storage tank is supplied to various components and areas, including the primary coolant head tank, pool surge tank, the primary and pool demineralizers, the beam room, and the TRU facility, by means of the two centrifugal demineralized water pumps, PU-18A and B. Each pump is rated at 100 gpm at 90 ft head with a maximum shutoff pressure of 150 psig. Power is supplied from the normal-power system, and the pumps must be restarted manually following recovery from a power outage.

10.2.4 Sprinkler System

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

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

To minimize accidental wetting of equipment, a preaction system is used in the reactor building. The sprinkler pipes are filled with compressed air and the water is contained behind a motor operated valve. This valve opens only upon a high-temperature indication from the fire alarm system; however, the fusable plugs must also melt before the water can be released from the sprinkler. This affords protection against a faulty sprinkler head or a false alarm. The system is self-supervised and annunciates on loss of air pressure or a failure of the fire alarm temperature indicator circuitry.

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.

10.3 Instrument Air System

Compressed air for instrument operation (Fig. 10.3.1) is normally furnished by three air compressors, C-1A, B, and C, each capable of delivering 100 scfm of completely oil-free air at pressures of 60 to 70 psig. Air from these compressors is stored in a receiver at 60 to 70 psig. The receiver pressure is controlled by a pressure switch and a compressor unloader valve. Two compressors are normally in standby while the other one is in operation. A standby emergency compressor (C-3) which has a capacity of 20 scfm is also provided for the purpose of supplying air to certain pneumatic instrumentation and control devices in the event of failure of the main compressors. Compressor C-1A receives normal power, C-1B receives power through normal-emergency system No. 1 and compressors C-1C and C-3 receive power through normal-emergency system No. 2. Not shown are the pressure-reducing valves which supply 20 or 30 psig air to the individual components.

The function of this air system is to provide motive power for the air-actuated controllers, motors, and instruments. Certain of these instruments are part of the control system and are necessary for all modes of operation. The heat power instrument air header is connected to the main air receiver and is supplied air by it during normal operation. The heat power instrument air header is also connected to emergency compressor C-3. A check valve in the main header permits air furnished by this compressor to supply only the heat power instrument header.

Should the main air receiver pressure drop below 53 psig, a pressure switch will start the standby compressors. The standby compressors will continue to run until manually stopped. If the heat power instrument header pressure drops below 43 psig, a pressure switch will start the emergency compressor. This compressor will also run until manually shut down.

Components supplied by the emergency air compressor include the following:

1. The letdown valves and the primary coolant pressure control instrumentation, which maintain primary system pressure
2. The secondary coolant throttling valves (TCV 377 and 377A) which control the water flow to the heat exchangers.
3. The primary heat exchanger inlet bypass valves, which permit pressure equalization when putting a standby heat exchanger into service.
4. Primary coolant flow and temperature transmitters and the various power computers and modifiers necessary to permit operation at N_L during a normal-power outage.

In addition to the instrument air system, two separate air compressors supply air for use in the dry-pipe fire protection sprinkler system at the cooling tower and the preaction system in the reactor building.

Air to drive hammers, vibrators, or other air-driven equipment of this type is furnished by a plant-air system, using installed piping and a portable compressor.

10.4 Alarm and Communications Systems

There are six alarm and communications systems at the HFIR site. 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

The locations of the various components of these systems are shown in Figs. 10.4.1–10.4.6.

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 HFIR area. Any reactor building fire alarm is also annunciated in the control room. In addition, fire alarm horns near the source of the alarm are energized.

Master box No. 1 serves the cooling tower, cooling tower equipment building, and the electrical building; master box No. 2 serves the entire reactor building; master box No. 3 serves the office and maintenance building; and master box No. 4 monitors for malfunction of the other three networks.

Heat-actuated devices operated either by temperature rate-of-rise or by high temperature are placed appropriately in all rooms. The maximum allowed separation 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 charger systems located in the HFIR 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 two master stations (one in the control room and the other in the reactor shift engineer's office) are capable of calling several stations at once to permit coordination of activity. The electrical power is supplied from normal-emergency system No. 1.

10.4.3 Sound Powered Phones

The sound powered phones are provided primarily for continuous communication between two areas for long periods of time, e.g., 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 calls. Power for the local switching system is provided by either the No. 1 or No. 2 normal-emergency system.

10.4.5 Public Address System

Microphones for the public address system are located in the reactor control room, supervisor's office, and in the office and maintenance building. Speakers are located in each of the major areas of the building. Additional speakers are located outside to serve the nearby area. Two amplifiers are used in this system – one is a standby. Power for the public address system is supplied from either the normal-emergency system No. 1 or No. 2.

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 HFIR local evacuation alarm, a tone signal obtained from an audio frequency generator on the PA amplifier rack, can be given by the control room operator. It may be actuated by switches located in the control room and in the reactor supervisor's office in the Office Maintenance Building. An automatic evacuation signal based on local radiation detection instrumentation, as described in Sec. 8.7.1, is installed.

11. SPECIAL SYSTEMS AND PROCEDURES

11.1 Gaseous Waste Disposal

Radioactive gaseous waste is disposed of by the two systems described in detail in Sec. 4: the special building hot exhaust (SBHE) and the hot off-gas (HOG) systems. The SBHE is a dynamic containment system intended to handle infrequent large-volume activity releases, whereas the HOG 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 250-ft HFIR 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 central air-conditioning system in such a way as to prevent in-leakage of air from areas of potential radioactive contamination as well as to prevent the spread of chemical fumes. 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 Dumpsters (10 × 5 × 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 Dumpsters 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 yellow 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 component under water and remove the pieces in lead casks. In the case of the SBHE and HOG filters, special shields are provided. The filters can be drawn into these shields, which are then removed by truck. In all cases, final disposal is accomplished by burial using existing ORNL equipment and facilities.[1]

11.3 Aqueous Waste Disposal

All the aqueous waste from the HFIR 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.

---

[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).

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

1. *Cold Waste.* 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 Melton Branch.

2. *Sewage.* This includes waste from showers, sinks, and toilet facilities. It is sent through the HFIR sewage treatment plant, and the effluent is discharged to Melton Branch.

3. *Process Waste Drainage (PWD).* 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 (~ 5700 dis min^{-1} ml^{-1}). The material is allowed to settle and to decay and then is gradually released to Melton Branch or pumped to the Melton Valley waste collection system.

4. *Intermediate Level Waste (ILW).* This originates as primary coolant leakage, demineralizer regeneration fluids, decontamination and "hot-sink" drainage and also includes 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. It is pumped to the ORNL waste disposal system.

The four aqueous waste disposal systems are shown in Fig. 8.7.2.

11.3.1 Process Waste Drainage

Process waste drainage from the SBHE filter pits, various floor drains and sinks, the primary coolant and pool cleanup systems, the pool surge tank, and the pool water storage tank flows directly by gravity to a PWD collection header. A 1000-gal sump is provided to collect waste from the subpile room, primary heat exchanger shell drains, and the SBHE drip drains which are located below the elevation at which the PWD header leaves the reactor building. This sump is emptied to the PWD header by two steam jets which are automatically operated by level switches. Each jet is designed to discharge 100 gpm against a 30-ft head.

The PWD collection header discharges directly into one of the two retention ponds located approximately 250 ft south of the reactor building. The smaller pond, No. 1, provides storage for 240,000 gal of liquid waste, which is estimated to be adequate to give a minimum retention of 12 hr under normal conditions. Normally, the process drainage wastes that enter this pond will contain no significant amounts of radioactivity. There is no special treatment; decontamination is by decay and settling. Waste enters the pond from the PWD header through a 24-in.-diam vitrified clay pipe in the north bank of the pond at elevation 798.5 ft, $2\frac{1}{2}$ ft above the pond bottom and $4\frac{1}{2}$ ft below the maximum high-water level.

Following retention, the waste leaves the pond through an 8-in.-diam 12-ft-long flexible hose which extends inside the pond. This hose is coupled to an 8-in. pipe which runs under the south embankment of the pond. The inner end of the hose may be raised and lowered by a winch in order to set the pond level. It may be lowered into a sump at elevation 796.5 ft to completely drain the pond. The effluent is pumped to the ORNL waste disposal system when the pond fills; provision exists for disposal to Melton Branch.

Retention pond No. 2 is used primarily to impound waste of higher activity concentration than that normally discharged to pond No. 1. This pond has a 500,000-gal capacity and is capable of impounding extreme flows which might result from the activation of a sprinkler system in a contaminated area. Flow from the PWD system can be diverted to pond No. 2 either manually or automatically when a high level of radiation is detected in the PWD line. Flow into and out of pond No. 2 is identical to that into and out of pond No. 1, except that the effluent stream can only be pumped to the ORNL waste disposal system.

The PWD header is equipped with a series of manholes and valve boxes which contain valves and measuring devices. Flow to the ponds is measured by a weir. A flow-proportional sampler monitors the radioactivity of the stream and retains a sample for analysis. Two diversion valves automatically divert the flow from pond No. 1 to pond No. 2 if a preset level of activity is exceeded. Two other diversion valves allow remote manual diversion of the flow to the ILW storage tank, if desired.

The cooling tower waste is primarily nonradioactive, chemically treated water and normally flows directly to Melton Branch. It may be diverted to pond No. 1 by automatically operated valves upon receipt of a high-radioactivity-level signal from a radiation monitor. Flow from the cooling tower is measured by a weir.

Each of the floor drains served by the PWD system has a trap which provides a water seal, a sediment bucket, and a cleanout. Floor drains in the reactor bay, experiment room, beam room, and process equipment rooms of the water wing are generally located near the columns. In other areas, such as the heating and ventilating and the electrical rooms, shielded compartments, and pipe tunnels, the drains are conveniently located with respect to the potential source of waste. A sump in the bottom of the elevator pit is pumped to the PWD system by a steam jet. Individual drain connections are located at each of the experiment stations, and a drain connection is located near the shielding wall at each beam port.

11.3.2 Intermediate-Level-Waste System

Intermediate level waste may be generated by various demineralizer operations, decontamination procedures, filter pits, or other equipment in the system. The wastes are collected at the point of origin by closed-pipe connections. Drain connections are located at the experiment service stations on the experiment area level and at the shield face on the beam room level. These drains are sealed except when placed in service, at which time they are connected to the equipment by welding. Connections are located in the reactor bay on each side of the reactor pool. Equipment drains, including the overflow from the primary system head tank, primary coolant loop, primary deaerator, spent resin disposal system, demineralizers, filters, and decontamination area drains, are connected to the ILW system. Wastes originating in the HOG systems are drained to a sump near the filter pits which drains to the ILW system. This sump also provides a connecting link through which the ILW system is vented to the open hot off-gas (OHOG) system.

The ILW lines are shielded with lead where they pass through open areas. The wastes from the various drains flow by gravity into the ILW collection header. This header leaves the reactor building at the southwest corner and conveys the waste to the ILW storage tank, which is located underground approximately 200-ft west of the building. This 8-ft-diam stainless steel tank has a capacity of 13,000 gal and is buried 19 ft underground. The liquid level in the tank is measured by a level sensor which provides a readout in the control room. Any overflow is drained to the PWD sump, where it can be diverted to the retention ponds.

The system is designed to deliver a maximum flow of 300 gpm to the tanks; however, this should seldom be achieved. Following sampling and analysis of the waste, the tank may be emptied by one or both of the two ILW transfer pumps, PU-4A and PU-4B. These self-priming pumps, each rated at 60 gpm at a 115-ft head, and associated piping are located in a shielded pit just north of the storage tank.

Waste from the tank can be transferred to the ORNL waste disposal system or to either of the HFIR retention ponds by a connection to the PWD header. The mode of disposal will depend upon the analysis of the tank contents. The pumps are also capable of recirculating the stored waste to prevent suspended solids from settling and to permit a representative sample to be obtained. A connection on the discharge side of each pump conveys a small portion of the waste to a sampling station. Drainage from this station is back to the ILW tank. A process water connection on the suction side of each pump provides clean water for flushing the pumps and outlet lines.

11.4 Resin Transfer System

To facilitate the removal of spent resin from the primary coolant and pool demineralizers, a special resin-removal line is connected to each of the demineralizer vessels. These lines originate just above the support screen for the resin and permit the resin to be removed hydraulically by flushing with water from demineralizer recycle pumps (see Figs. 6.2.11 and 6.4.5).

Resin from each of the six demineralizer beds is flushed to a common measuring tank located in a pit on the north side of the building. The water is separated from the slurry by a screen on the bottom of the tank and is recycled to the demineralizer. Process water is used to sluice the resin from the measuring tank into disposable drums. Screens in the bottom of these drums permit the sluice water to drain to the pit and into the ILW system. The drums are then loaded in trucks and disposed of by burial.

Normally the resin is regenerated at least once prior to removal to reduce its radioactivity. The pit is shielded by 2 ft of concrete during the flushing and sluicing procedures, which may be done remotely; however, the transfer of the drum from the pit is normally an unshielded operation; however, special precautions are taken if the waste is highly radioactive.

11.5 Decontamination Facilities

A local equipment decontamination facility is provided at the HFIR site in the northwest corner of the reactor bay (see Fig. 3.2.3). This facility is a 22- by 22-ft stainless steel floored area provided with a floor drain connected to the PWD system. A 6-in. curb prevents water or other decontamination agents from reaching the rest of the bay floor. Services installed include a 3- by 4-ft stainless steel sink, a 13- by 13-in. SBHE duct outlet, a 3-in. ILW connection, a 2-in. OHOG connection, a 2-in. process water connection, and the 3-in. PWD floor drain. Pipe sockets are installed in the curbing so the area may be roped off. The design floor loading is adequate to support a 50-ton shipping cask.

Room 207, the contaminated clothing room, is intended for use as a personnel decontamination facility as well as an air lock. Personnel using this room can enter it directly from the reactor bay, remove contaminated clothing, shower, dress, and leave by the door leading to the clean area. Room 207 has poured concrete walls 8 in. thick and is located opposite the Health Physics office. Normally, Room 207 is cooled by the central air-conditioning unit AC-15 and is arranged to provide for a slight inward leakage from the clean area. Pressure differences between Room 207 and the reactor bay provide air leakage from the room into the reactor bay. A 9-in.-long removable section in both the air-conditioning supply and return ducts may be removed to isolate the room when it is contaminated.

11.6 Fuel Storage and Handling

The HFIR fuel cores consist of two concentric fuel elements containing a total of 9.4 kg of U^{235} in the form of 93% enriched uranium. In general, the fuel elements are handled separately, but the cores are stored as units. The cores will be received from the vendor, inspected, and then stored on individual skids in a vault located in the southeast corner of the first floor of the reactor building (Fig. 3.2.2). From here they will be taken as needed to the reactor bay by means of a manually operated fork lift. The skids are so designed that the collection of stored fuel elements forms an "always-safe" array.

Spent fuel cores are stored under water in the storage pools, as indicated in Figs. 3.1.1 and 9.6.1. Twenty-seven horizontal racks, each of which is provided with a hollow cadmium-filled post and a cadmium-filled cylinder, are furnished for this purpose. A typical rack is shown in Fig. 9.6.2. When a spent fuel core has decayed sufficiently, it is loaded under water into a lead-shielded cask and shipped to the fuel reprocessing plant.

Two defective fuel element storage tanks are located in the reactor pool (Fig. 9.6.3). The purpose of these tanks is to provide storage for fuel elements which have cladding leaks during the time required for short-lived fission product decay. Pool water from the defective fuel element storage tanks flows directly to the pool cleanup system, thus minimizing contamination of the reactor pool with fission products from a leaking element.

The defective fuel element storage tanks are provided with hollow cadmium-filled posts and cylinders similar to those of the spent fuel racks. In addition, the tanks are so arranged that the cores are cooled by a convection loop: should flow to the pool cleanup system be blocked, cooling is accomplished by natural convection through a 300-kw heat exchanger cooled by pool water.

All procedures used to handle HFIR fuel are governed by the necessity to prevent inadvertent criticality, overheating of spent elements, and excessive radiation exposure. In addition, precautions are taken to prevent excessive gamma heating in the concrete structure by freshly removed spent fuel.

11.7 Critical Facility

Provisions have been made at the east end of the storage pool to permit the future installation of a critical facility which will use HFIR fuel elements. A circular pool 8 ft in diameter and 25 ft deep will accommodate the facility. Appropriate services and conduit are available at this location. Space for the control console and instruments is provided in the HFIR reactor control room.

11.8 Testing of Process Instruments for Safety Channels

As described in Sec. 8.4.2, each sensor associated with the safety channels may be tested during operation by locally perturbing the parameter measured by that one sensor and observing the behavior of the safety plates' magnet current.

Testing of the ion chambers is described in Sec. 8.5.2(*f*). The primary coolant pressure sensors are tested by opening a small valve (HCV 128A1, -B1, or -C1) in the instrument tubing associated with the pressure transducer. This allows the water in the tubing to flow to the ILW drains; the resultant lower pressure causes a scram trip in that channel. The flow sensor is tested similarly. Since the flow signal is obtained by measuring the pressure difference across a venturi, opening a valve which connects the two instrument tubing leads equalizes the pressure and causes a scram trip.

The temperature sensors are tested by injecting heated primary-coolant-cleanup return water into the primary coolant flow so that the high-temperature water strikes the temperature sensor. The temperature sensors are mounted in a flange (Fig. 11.8.1), with a special water injection nozzle. Various control valves, operated from the control console, direct the flow against the chosen sensor (Fig. 11.8.2). The high-pressure cleanup return flow is momentarily throttled during testing, diverting part of its flow through the heat exchanger to provide the high pressure necessary to force the hot water into the primary coolant flow.

The primary coolant gross radiation monitor system is tested by removing a lead shield from in front of each of the three ion chambers to test each of the channels.

11.9 Poison Injection System

A 100-gal tank (Fig. 6.2.11), containing 44 wt % of cadmium nitrate in water, may be drained in ~ 100 sec into the suction of the pressurizer pumps upon indication that the four shim-safety plates and the shim-regulating cylinder are incapable of insertion.

12. ORGANIZATION AND ADMINISTRATION

12.1 Introduction

The HFIR 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 vigorously engaged in the solution of problems pertaining to almost every aspect of the atomic energy program, particularly those concerned with the peaceful application of atomic energy. There are nearly 4800 employees in some 100 research groups at work in eight 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 production and 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 evaluating the potential hazards they may introduce, and to controlling radiation and radioactive materials so that people will not be exposed to quantities 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 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, investigation into the 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 HFIR, 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 yr. 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 core, 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 x 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 pre-test elements for the ORR.

The Oak Ridge Research Reactor is a natural water moderated and cooled, beryllium reflected, 30 Mw research and testing reactor using highly enriched uranium in its core. It first achieved criticality on March 21, 1958. The ORR produces most of the radioisotopes which are produced by the Laboratory.

The Tower Shielding Facility (TSF), 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 U^{235} 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 10-Mw (th) Molten-Salt Reactor Experiment is demonstrating 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 nuclear propulsion.

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 a 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 Institute of Nuclear Studies. 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 masters or doctoral degrees at the Laboratory through AEC fellowships and other arrangements. ORNL also operated the Oak Ridge School of Reactor Technology (ORSORT) for training graduate nuclear engineers.

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. 12.2.1.

Operation of the HFIR is the direct responsibility of the Operations Division, which is also responsible for the Oak Ridge Research Reactor (ORR), the Low Intensity Test 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. 12.2.2.

In addition to the Operations Division personnel, who perform both day-to-day operation of the three reactors and 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 is responsible not only 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, and, when needed, additional personnel are made available from the parent organizations.

The HFIR organization itself is shown in Fig. 12.2.3. The normal paths of communication with the service groups are indicated by dotted lines.

The personnel[1] of the Operations Division may be divided broadly into three categories:

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. He must have 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 will include an undergraduate degree in engineering or one of the physical sciences plus considerable specialized work in reactor technology. The senior reactor supervision and the technical support personnel of the Division fall into this category.

The operation 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 training of the operation engineer emphasizes familiarity with the reactor and other devices under his control and their behavior and, in particular, their limitations. The shift engineers are included in this classification. Usually they are required to have an undergraduate degree in engineering and some specialized training, often acquired on the job, in nuclear engineering.

The reactor operators are the individuals who, under the supervision of the reactor engineer or the operating 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, manually dextrous, and have a reasonable degree of intelligence. The educational requirement is a high school diploma or equivalent.

12.3 Training and Qualification of Operations Department Personnel

All of the engineers currently employed by the Division have undergraduate degrees in engineering or in one of the physical sciences, and they have an average of 5 yr of reactor operating experience. Thus there is a reservoir of trained supervisory personnel which will be utilized for the operation of the HFIR. Those engineers initially assigned to the HFIR received additional training designed to thoroughly familiarize them with that facility. In part this took the form of following the design and construction of the facility and of assisting in the preparation of test, start-up, and operating procedures.

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 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 Chapter 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 independently carry out his tasks in a competent manner.

Training of the operating engineer takes somewhat longer, and it generally requires at least six months 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.

---

[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 Held in Vienna, Austria*, Academic Press, New York, 1962.

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 HFIR, the Reactor Supervisor has 13 yr of reactor operating experience; his assistant, an ORSORT graduate, 8 yr. The two day-shift engineers have at least 3 yr of experience, and the shift engineers have a minimum of 2 yr of experience. The operators initially assigned received several months training at the HFIR, and in many cases had extensive operating experience at the ORR or the LITR.

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 six reactor engineers who are responsible for long-range improvements in operating techniques and safety. The personnel of this group, which is responsible to the Division Superintendent, are all graduate engineers with either long experience in the reactor engineering field, 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 six 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 ORNL 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 and which are listed together with a brief resume 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 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 P and E 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, including the present HFIR fuel element.

The P and E 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. This is particularly true for the HFIR, for which the Division has performed critical experiments. 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 on irradiated material. Many radiation damage experiments and studies on defects in general often require radiation sources other than reactors, and the division possesses several Co^{60} gamma sources and Cs^{137} 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; 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 extra-code 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 in ORNL; recommends repairs and reinspects after repair; estimates and recommends retirement dates for deteriorated equipment; 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; carries 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; includes most of the chemists formerly engaged in chemical studies of the aqueous homogeneous reactor program; hence can furnish background information on the behavior of high temperature aqueous systems; includes 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; and has expert personnel and facilities for identification of particulate material or other solids found in the course of reactor operation.

The Health Physics Division is responsible for personnel monitoring and maintaining personnel exposure records; is responsible for radiation monitoring of all operations, 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; 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 pilot 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 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 close 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 originate 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 are responsible for the mechanical and nuclear design of HFIR.

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 approximately 36 engineers and 30 draftsmen.

The *Analysis Department* is composed of 22 professional people, including seven 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 in reactor safety analysis.

12.5 Method of Operation

As is the case with the other reactors operated by the Operations Division, the HFIR is operated through the use of carefully prepared, written standard procedures, 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 manual* for the HFIR contains material covering the following subjects:

Startup

Steady-State Power Operation

Shutdowns

**Operating Manual for the HFIR — Operating Procedures*, ORNL-TM-1168 (July 1965).

Instruments and Controls

Storage, Refueling, and Pool Work

In-Core Work

Replacements of Components

Replacement of Experimental Facility Tubes

Research

Reactor Primary Coolant System

Reactor Cleanup System

Pool Cooling System

Pool Cleanup System

Secondary Cooling System

Maintenance

Containment; Heating, Ventilating, and Air Conditioning

Emergency Procedures and Evacuation

Radioactive Waste Systems

Radiation Safety and Control

Records and Data Accumulation

Initially, the operating procedures are written by Operations Division personnel, in cooperation with members of the project design group, 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, daily shift checks, as well as 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, will be 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 HFIR 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
2. Shutdowns
3. Trouble
4. Maintenance
5. Service to Research
6. Routine Checks
7. Sample Irradiations
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 instrument and control complement and with respect to routine lubrication and maintenance of mechanical equipment. In critical cases, instrument test procedures are supplemented by the use of a checklist and may be considered a part of the operating procedure. An IBM-card system is utilized to keep abreast of routine mechanical maintenance and the stocking of spare parts.

12.6 Testing and Startup Procedures

Procedures for component and system testing were prepared both by the architect-engineer and by ORNL personnel. In general, the acceptance testing of the conventional equipment was specified by the architect-engineer, and testing of the more specialized equipment closely related to reactor operation was specified by ORNL. All the tests were witnessed and approved by ORNL personnel.

Testing of the various reactor systems was conducted primarily by the Operations Division, the Plant and Equipment Division, and the HFIR Project. Considerable attention was given to the study and demonstration of the methods of assembling and disassembling the core components, as well as to the investigation of the hydraulic and mechanical characteristics of the various water systems and the behavior of the dynamic containment.

The startup and postneutron test procedures included a critical experiment in the HFIR utilizing the same core and control plates which were extensively investigated in the HFIRCE-3 critical experiment program. This series of tests was performed to verify the similarity of the HFIRCE-3 assembly with the reactor. In addition, various experiments were planned and conducted to give the operators a degree of familiarity with the reactor behavior while working with a core of known characteristics.

Following the series of tests with the HFIRCE-3 core and control plates, an orderly transition was made to the first production core loading with production control plates. A discussion of the several cores and control plates is given in Section II, D of Appendix C, ORNL-3573. This series of tests included verification of shutdown margin, instrument behavior at measurable power levels, heat transfer adequacy for low-power operation, etc. The initial critical experiments were started in August 1965, with initial criticality being reached at 2:22 p.m., August 25, 1965. Full-power operation at 100 Mw was begun on September 9, 1966, after a series of stepwise increases in power level starting at 20 Mw on January 29, 1966.

12.7 Internal Safety Reviews

Aside from the intradepartmental 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, four will be concerned with the HFIR 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 in the reactor.

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 for insertion in the reactor by the Technical Assistance Department. 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 in 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.

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 of 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.

---

[2] Francois Kertesz, *The Auditing of Reactor Safety at the Oak Ridge National Laboratory*, ORNL-TM-612 (July 8, 1963).

[3] C. D. Cagle, *General Standards Guide for Experiments in ORNL Research Reactors*, ORNL-TM-281 (Aug. 20, 1962).

The Waste Effluents Review Committee audits the practices used in waste disposal at the Laboratory. It examines in detail the various procedures in use to dispose of liquid, solid, and gaseous waste from the reactor to ensure that no hazard to the public or to Laboratory personnel will ensue. From time to time, this committee may require safety analyses of various processes in order to implement its function.

In addition to these standing committees an HFIR Design Review Committee was established by the Laboratory Director. The function of this committee, composed of 13 senior staff members of the Laboratory, was to review the HFIR design for safety and feasibility throughout the design period. The committee itself did not participate in the design but limited its activities to an evaluation of the proposals presented by the designers.

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, 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 following closely 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, reactor safety, etc., are on the staff of the Director of Safety and Radiation Control.

Appendix A

HFIR DESIGN PARAMETERS

Values tabulated for parameters are the nominal values.

I. Reactor Power Levels, Mw	
A. Steady state, operating	100
B. Reverse point	110
C. Scram point (neutron flux)	130
D. Scram point (heat power)	120
E. Minimum steady-state incipient boiling	142
II. Neutron Fluxes at 100 Mw, neutrons cm^{-2} sec^{-1}	
A. Thermal	
1. Maximum unperturbed in island	5.5×10^{15}
2. Average in typical 300-g Pu^{242} island target	2.0×10^{15}
3. Maximum unperturbed in Be reflector	
a. Beginning of fuel cycle	1.1×10^{15}
b. End of fuel cycle	1.6×10^{15}
4. Maximum unperturbed at Be-H_2O reflector interface	
a. Beginning of fuel cycle	1.4×10^{14}
b. End of fuel cycle	1.7×10^{14}
5. Average in fuel region	
a. Beginning of cycle	3.3×10^{14}
b. End of fuel cycle	4.5×10^{14}
B. Total nonthermal	
1. Average in island target	2.4×10^{15}
2. Maximum in fuel region	4.0×10^{15}
III. Reactor Materials	
A. Fuel plate	U_3O_8-Al cermet, Al cladding
1. Weight of U^{235} per plate in inner fuel element, g	15.18 ± 1%
2. Weight of U^{235} per plate in outer fuel element, g	18.44 ± 1%
3. U^{235} enrichment	~93%
4. Total fuel loading of U^{235}, kg	9.40

5. Total burnable poison loading (B^{10} in inner fuel element plates only), g	2.8
B. Coolant	H_2O
C. Island moderator	H_2O
D. Side reflector	
Removable	Be + 5% H_2O
Permanent	Be + 2% H_2O
E. Shim, safety and regulating plates	
(Black region)	Eu_2O_3 + Al
(Gray region)	Ta + Al
(White region)	Al
F. Plutonium target rods	PuO_2-Al cermet, Al cladding
1. Total loading of Pu^{242} in typical target, g	265
IV. Heat Transfer and Coolant Data	
A. General	
1. Design heat loads (at 100 Mw steady-state reactor power), Mw	
a. Fuel element	97.5
b. Target	0.90
c. Control rods	4.7
d. Beryllium reflector	3.7
2. System design pressure, psi	1000
3. Normal operating pressure (vessel inlet), psi	600
4. Design coolant flow rates,* gpm	
a. Total	~16,000
b. Through fuel channels	~13,000
c. Between fuel elements	133
d. Through island region containing typical target	788
e. Beryllium reflector	740
f. Control plate region	1080
g. Experimental facilities	365
5. System pressure drop, psi	~156
6. Coolant temperatures (at 100 Mw), °F	
a. Vessel inlet	120
b. Vessel outlet	164
c. Maximum bulk water (fuel outlet)	249
d. Maximum surface (fuel plate–water interface)	387

*All flow rates, except total and fuel channel, were obtained by suitable orificing following flow tests *in situ*. Total and fuel channel flow rates given here are nominal flow rates.

B. Fuel region

1. Geometry: Two concentric cylindrical annuli, involute fuel plates	
2. Fuel element dimensions	
a. Inner fuel element dimeters, in.	
(1) ID	5.067
(2) OD	10.590
b. Outer fuel element diameters, in.	
(1) ID	11.250
(2) OD	17.134
c. Active fuel region diameters, in. (avg.)	
(1) Inner fuel element, ID	5.623
(2) Inner fuel element, OD	9.920
(3) Outer fuel element, ID	11.913
(4) Outer fuel element, OD	16.483
d. Height of active core, in.	20
e. Total fuel plate height, in.	24
f. Fuel plate thickness, in.	0.050
g. Cladding thickness (each surface), in.	0.010
h. Fuel-plate core thickness, in.	0.030
i. Coolant channel thickness, in.	0.050
3. Number of fuel plates	
a. Inner fuel element	171
b. Outer fuel element	369
4. Total heat transfer area, ft^2 (avg.)	428.8
5. Volume of active core, liters (avg.)	50.59
6. Heat load (at 100 Mw steady-state reactor power), Mw	97.5
7. Power density, Mw/liter	
a. Maximum	4.38
b. Average	1.93
8. Heat flux, Btu hr^{-1} ft^{-2}	
a. Hot spot	1.97×10^6
b. Average	0.776×10^6
c. Hot spot, at incipient boiling power level (600 psi nominal operating pressure) at beginning of fuel cycle	3.4×10^6
9. Coolant flow rate, gpm	$\sim$13,000
10. Coolant velocity, ft/sec	51
11. Pressure drop across fuel element, psi	$\sim$108
12. Temperature, °F	
a. Inlet	120

b. Outlet (nominal)	163–196
c. Oxide-water interface (max)	387
e. Oxide-water interface (nominal)	146–269
e. Metal-oxide interface (max)	589
f. Metal-oxide interface (nominal)	149–303
g. Fuel plate (max)	620
h. Fuel plate (nominal)	153–325
i. Saturation temperature at hot spot	465

C. Target region

1. Geometry: Cylindrical target rods spaced on triangular pattern

2. Number of rods	31
3. Target region dimensions	
a. Diameter of water island, in.	5.067
b. Rod diameter, in.	0.360
c. Height of active portion, in.	20
d. Total rod length, in.	35
e. Spacing between rod centers, in.	0.665
4. Heat load, kw (max)	900
5. Heat flux (av), Btu hr^{-1} ft^{-2}	0.6×10^6
6. Water flow rate, gpm	788
7. Nominal water velocity around rods, ft/sec	40
8. Pressure drop across sample array, psi	45
9. Temperatures, °F	
a. Water inlet	120
b. Water outlet	129
c. Rod oxide-film surface (max)	255

D. Control region

1. Geometry: Annular region located between fuel region and beryllium reflector; contains five control plates (one regulating-shim, four shim-safeties) which constitute two concentric cylinders.

2. Control region dimensions	
a. Overall region radii, in.	
(1) Inside	8.567
(2) Outside	9.436
b. Control plate radii, in.	
(1) Inner control cylinder, inside	8.671
(2) Inner control cylinder, outside	8.921
(3) Outer control quadrants, inside	9.050
(4) Outer control quadrants, outside	9.300

(5) Height within core region, in.

Black region	22
Gray region	5
White region	~20

c. Coolant channel thickness, in. (starting with the inner channel) — 0.104, 0.170, 0.095

3. Heat load, Mw	4.7
4. Maximum heat generation rate, watts/g	42
5. Heat flux (max), Btu hr^{-1} ft^{-2}	380,000
6. Water flow rate, gpm	1080
7. Water velocity (nominal), fps	16
8. Pressure drop across control plates, psi	~9
9. Temperatures, °F	
a. Water inlet	120
b. Water outlet (av)	137
c. Cylinder surface (max)	205

E. Removable beryllium reflector

1. Geometry: Three concentric cylinders cooled with water flowing through $\frac{1}{8}$-in. axial grooves on the cylinder surface; inner cylinder lined with aluminum.

2. Removable beryllium dimensions, in.

a. Overall ID	18.872
b. Overall OD	23.756
c. Height	24.0

d. Nominal radii of individual cylinders, in.

	Material	Inside	Outside
No. 1	Al	9.436	9.498
No. 2	Be	9.498	9.889
No. 3	Be	9.889	10.757
No. 4	Be	10.757	11.878

3. Heat load, Mw	2
4. Maximum heat generation rate, watts/g of Be	31
5. Heat flux (max), Btu hr^{-1} ft^{-2}	0.2×10^6
6. Coolant velocity, ft/sec	12.5
7. Total water flow rate, gpm	175
8. Pressure drop, psi	~9
9. Temperatures, °F	
a. Water inlet	120
b. Water outlet (nominal)	166
c. Beryllium surface (max)	185
d. Beryllium interior (max)	263

F. Permanent and semipermanent beryllium reflector (including control plate access plugs)

1. Geometry: Beryllium annulus with axial coolant holes, divided into semipermanent and permanent beryllium at inner circle of coolant holes.

2. Beryllium dimensions, in.

a. ID (semipermanent reflector)	23.884
b. ID (interface)	26.250
c. OD (permanent reflector)	43.0
d. Height	24.0
e. Coolant hole diameter	0.125

f. Coolant hole spacing: Located on 5 concentric circles, 80 uniformly spaced holes per circle except where altered by experimental facilities

g. Radii of coolant hole circles, in.

Circle No. 1	13.08
Circle No. 2	14.124
Circle No. 3	15.306
Circle No. 4	16.662
Circle No. 5	18.219

3. Heat load, Mw	1.7
4. Maximum heat generation rate, watts/g of Be	11.5
5. Heat flux (max), Btu hr^{-1} ft^{-2}	0.3×10^6
6. Total water flow rate, gpm	565
7. Coolant velocity, ft/sec	13
8. Temperatures, °F	
a. Water inlet	120
b. Water outlet (av from $\frac{1}{8}$-in. coolant holes)	134
c. Beryllium surface (max)	230
d. Interior beryllium (max)	290

V. Experiment Facilities

A. Permanent reflector experimental facilities

1. 1.584-in.-diam vertical facilities

a. Number	16
b. Location (distance from core centerline), in.	
(1) 11 holes	15.4375
(2) 5 holes	17.344

2. 2.834-in.-diam vertical facilities

a. Number	6
b. Location (distance from core centerline), in.	18.219

3. Horizontal beam holes
   a. Number — 3
   b. Location: Beam centerline in core horizontal midplane. One beam radial to active core; one beam tangent to active core; and one beam penetrating entire reflector-vessel assembly.
4. Engineering facilities
   a. Number — 4
   b. Location: At outer periphery of beryllium reflector, oriented at 41° angle to the vertical, and recessed a maximum of $2\frac{1}{4}$ in. into the permanent beryllium.
5. $\frac{1}{2}$-in.-diam vertical facilities
   a. Number — 8
   b. Location: Two in each shim plate access plug

B. Removable Reflector Experimental Facilities
1. $\frac{1}{2}$-in.-diam vertical facilities
   a. Number — 8
   b. Location: Removable beryllium Cylinder No. 3

VI. Reactivities

A. Summary of reactivity accountability for 9.4-kg HFIR core (2.8 g of B^{10})

Parameter	Reactivities ($\Delta k/k$) Time in Cycle (days) 0	2	14
1. Fuel worth for following conditions (zero power, 70°F, no boron burnable poison, no target, no beryllium poison):	0.135		
2. Temperature deficit (with 310 g Pu target, zero power at 70°F to 100 Mw)	–0.004		
3. Boron burnable poison	–0.05	–0.037	–0.009
4. Plutonium target (300 g $Pu^{242}O_2$ + 3 g $Pu^{241}O_2$ + 3 g $Pu^{239}O_2$)			
a. Time zero (max)	+0.008		
b. 0.1 yr (min)	–0.001		
c. 0.4 yr (max)	+0.008		
5. Xe^{135} + Sm^{149}	0	–0.049	–0.053
6. All fission products	0	–0.053	–0.086
7. Be poison (Li^6 + He^3)			
a. Time zero	0		
b. 0.2 yr	–0.013		
c. 5 yr	–0.016		

8. Beam tube flooding ~0
9. Fuel loading tolerance (±1%) ±.0015
10. Boron loading tolerance (±10%) ±.0038
11. Fuel distribution tolerance (± 10%) ±.0054
12. Boron distribution tolerance (±35%) ±.0023
13. Minimum $k_{eff} - 1$ (clean core, 100 Mw) 0.054
14. Maximum $k_{eff} - 1$ (clean core, 70°F) 0.106
15. Control plate worths
    a. All plates inserted 0.187
    b. Inner cylinder withdrawn, 4 quadrants inserted 0.147
16. Shutdown margins for maximum k_{eff} case
    a. All plates inserted 0.069
    b. Inner cylinder withdrawn, 4 quadrants inserted 0.030
17. Shutdown margins for typical nominal k_{eff} case
    a. All plates inserted 0.081
    b. Inner cylinder withdrawn, 4 quadrants inserted 0.042

B. Control plate characteristics

1. Control plate total worths, $\Delta k/k$
    a. All plates inserted 0.187
    b. Inner cylinder withdrawn, 4 quadrants inserted 0.147
2. Maximum differential plate worth, $\Delta k/k$ per in. (elements symmetrical)
    a. Inner cylinder (regulating-shim) 0.011
    b. Four outer quadrants (shim-safeties) ganged 0.010
3. Control plate speeds, in./sec
    a. Outer plate shim action 0.0958
    b. Outer plate fast rundown 1.00
    c. Inner plate shim action 0.0958
    d. Inner plate regulating action (max) 0.250
4. Control plate maximum reactivity insertion rates, $\Delta k/k$ per sec
    a. Outer plate shim action (four quadrants ganged) ±0.00096
    b. Outer plate fast rundown (four quadrants ganged) –0.010
    c. Inner cylinder shim action ±0.0011
    d. Inner cylinder regulating action (max) ±0.0028

C. Reactivity coefficients (see Sec. 7.3.4)

1. Temperature coefficients of reactivity for the clean core

Condition	$\Delta k/k$ (°F)$^{-1}$ at 79°F	$\Delta k/k$ (°F)$^{-1}$ at 155°F
With 310 g PuO_2 target in island		
Overall isothermal	-1.16×10^{-5}	-3.1×10^{-5}
Fuel region only	-7.3×10^{-5}	-10.2×10^{-5}
Island + control + reflector regions (by difference)	$+6.1 \times 10^{-5}$	$+7.1 \times 10^{-5}$
Without target in island		
Overall isothermal	$+2.3 \times 10^{-5}$	-2.1×10^{-5}
Fuel region only	-6.0×10^{-5}	-11.7×10^{-5}
Island + control + reflector regions (by difference)	$+8.3 \times 10^{-5}$	$+9.6 \times 10^{-5}$

2. Void reactivity coefficients (uniform voids), $(\Delta k/k)/(\Delta V/V)$

a. No target in island

(1) Island ($<$20% voids)	+0.06
($\sim$70% voids)	0
Max positive change in k =	$0.032\Delta k$
(2) Fuel region	
Inner fuel element	−0.188
Outer fuel element	−0.384
Annulus between fuel elements	−0.046

b. With typical target in island

(1) Island ($<$20% voids)	+0.05
($\sim$42% voids)	0
Max positive change in k =	$0.016\Delta k$
(2) Fuel region (same as without target)	

D. Miscellaneous

1. Xenon changes	100% to 10% power
a. Time to zero reactivity at 10 days	7.5 min
b. Time to zero reactivity at 1.0 days	35 min
2. Scram times, sec	
a. Latch release	$<$0.010
b. Plate movement, 6 in.	$\sim$0.090
c. Plate movement, full travel	$\sim$0.280
3. Prompt neutron lifetime (μsec)	
a. Beginning of cycle	35
b. 10 days	70

Appendix B

MANUFACTURERS SPECIFICATIONS

Reactor Pressure Vessel 01-RV-1

Manufacturer Allis-Chalmers Manufacturing Company
Nuclear Power Department
Milwaukee, Wisconsin

General Description. – 94-in.-diam carbon steel vessel. Carbon steel thickness $2\frac{7}{8}$ in., with a minimum of $\frac{1}{8}$-in.- thick austenitic stainless steel inside and 0.10 in. outside. A solid stainless steel extension, with an integral barytes concrete outer shield plug, is attached to the lower end of the vessel to permit control rod access to the core through the 7-ft-thick pool floor.

The following nozzles have been provided:

Nomenclature	Purpose	Number	Size	Elevation from Core Midplane
	Cooling water inlet	2	16″ OD pipe	5 ft 6 in. above
	Cooling water exit	1	18″ OD pipe	7 ft 0 in. below
EF	Slant engineering facility penetration	4	6″ 600 lb RJ flange	5 ft 9 in. above
IC	Ion chamber penetration	3	4″ 600 lb RJ flange	10 ft $\frac{1}{8}$ in. above
RH	Rabbit hole penetration	8	3″ 600 lb RJ flange	3 ft 6 in. above
HB No. 2	Radial horizontal beam tube penetration	1	14″ 600 lb RJ flange	Midplane
HB 1, 3, 4	Tangential horizontal beam tube penetration	3	8″ 600 lb RJ flange	Midplane
VS	Vertical experimental penetration	12	2″ 600 lb RJ flange	Top head
VL	Vertical experimental penetration	9	4″ 600 lb RJ flange	Top head
Lower extension	Control rod access	1	30″ flange	15 ft 6 in. below

The following closures have been provided:

Nomenclature	Purpose	Closure Type	Seal Type	Location
Top head	To provide unimpeded access and visibility to the entire ID of the vessel for maintenance, experiment installation, and decontamination	Flat, bolted head	Elastomer O-ring	Top is 11 ft 1 in. above core mid-plane
Quick opening hatch	To provide a 30-in.-diam clear hole in top head for normal core replacement and to permit installation and removal of removable reflector assembly	Breech lock	Multiple piston type elastomer O-rings	Center of top
Target hole plug	To provide a 12-in.-diam separately removable plug for possible target instrumentation penetrations and a rabbit tube	Shear block	Multiple piston type elastomer O-rings	Center of quick opening hatch
Lower head	Original installation inspection	Flat, bolted head	Elastomer O-ring	Parting line 15 ft 9 in. below core midplane
All other nozzles	Experiment or beam tube installation and removal	ASA bolted flanged joints	Metallic O-rings	

Materials of Construction

Item	Material
Shell and hemispherical head	ASTM A212 grade B plate clad on inside with 304L stainless steel plate and on outside with 347 stainless steel weld overlay
HB 1 and 4 nozzles	ASTM A105 grade II forgings clad overall with 347 stainless steel weld overlay
RH, EF, IC, and coolant water inlet nozzles	ASTM A105 grade II forgings clad overall with 347 stainless steel weld overlay
Top head and flange	ASTM A105 grade II forgings clad overall with 347 stainless steel weld overlay
HB No. 2 nozzle	ASTM A350 grade LF3 forgings clad with 347 stainless steel weld overlay
HB No. 3 nozzle	ASTM A350 grade LF3 forgings clad with 347 stainless steel weld overlay

Materials of Construction (continued)

Item	Material
Lower extension and all nozzle extensions	ASTM A240, A182, or A336, 304L stainless steel
Quick-opening hatch, target hole plug, and lower head	ASTM A182 or A336 304 stainless steel
Top and bottom head bolting	ASTM A193 grade B6 type 416 stainless steel with special corrosion protection film for top head bolting
Target hole plug shear blocks	Inconel
Nozzle bolting	ASTM A193 grade B8F-303
Main closure gaskets (top and bottom heads, quick-opening hatch, and target hole plug)	Hycar nitrile antirad rubber
Other gaskets	304 stainless steel
Concrete for shield plug	Barytes
Dissimilar metal welds	Inconel

Design Criteria. – Design criteria for the vessel are contained in ORNL Specification RD-8.8-1 and ORNL Drawings TD-E-5595, TD-E-5607, TD-D-5613, and TD-E-5614.

Design Conditions. – The vessel has been designed in accordance with Sect. VIII of the ASME Boiler and Pressure Vessel Code and Nuclear Code Cases 1270N, 1271N, and 1273N. Structural analysis has been performed by the methods described in the Department of Commerce Bulletin, PB 151987, "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components," December 1958 revision.

The loading conditions which have been evaluated are outlined below.

Design Conditions:

	Operating	Design	Test
Temperature (°F)	120-167	200	~75
Pressure (psig)	600	1000	1550

Normal Operating Conditions (2000 cycles)

	Primary Coolant		Pool	Gamma Heating (watts/gm)
	Temperature (°F)	Pressure (psig)	Temperature (°F)	
Vessel inlet	120	1000	105	0.05
Vessel exit	167	925	95	0.05

Design life is 20 years.

Test and Inspection. – The vessel was inspected in accordance with the requirements of the Code and Nuclear Code Case 1273N. Additional inspections performed included penetrant and magnetic particle (as applicable), inspection of all welding including overlay cladding both before and after stress relief, penetrant inspection of bolting threads, impact testing of ferritic materials, and ultrasonic testing of plates, forgings, and other vessel parts. All inspections were witnessed by representatives of the ORNL Inspection Engineering Department. Complete inspection records are maintained by the Allis-Chalmers Non-Destructive Test Group at West Allis, Wisconsin and by the ORNL Inspection Engineering Department.

The following tests to verify the adequacy of the vessel have been performed:

1. Hydrostatic test at 1550 psig.
2. Additional hydrostatic leak tests at 1000 psig following 17 cycles from atmospheric pressure to 1000 psig.
3. Measurement of top head bolt tensile stress under preload and under 1550 psig and 1000 psig hydrostatic heads will be made.

Primary Coolant Piping (High Pressure Supply and Return)

Prefabricated by Southwest Fabricating Welding Company
P. O. Box 9175
Central Park Station
Houston 11, Texas

Installed by The H. K. Ferguson Company

General Description. – All piping 304 stainless steel. Design pressure 1000 psi at 200°F. Joint connections welded and flanged, with 100% welding inspection by radiographic and liquid-penetrant techniques.

All piping complies with the requirement in the following codes and standards:

(1) Code for Pressure Piping, ASA-B31.1 (1955), Sect. 1 and 6 and nuclear case interpretations N-1, N-3, N-5, N-7, and N-10

(2) Plate, sheet and strip – ASTM A-240

(3) Pipe, 6 in. and smaller – ASA B36-10 (1959), ASTM A-376

(4) Pipe, 8 in. through 18 in. – ASA B36.10 (1959), ASTM A-358

(5) Pipe, 20 in. – ASA B36.10 (1959), ASTM A-358, A-240

(6) Fittings – ASA B16.9 (1958)

(7) Bolts, chrome-moly steel stud, threaded full length – ASTM A-193-60T, Grade B-7

(8) Nuts, carbon steel – ASTM-A-194-59T, Grade 2-H

(9) Welder qualification – ASME Section 9

Primary Coolant Heat Exchanger 01-EX-1A, -1B, -1C, -1D

Manufacturer

A. O. Smith Corporation
Process Equipment Division
Milwaukee, Wisconsin

Manufacturers Size 48-344, Type CGU (Vertical Channel)

Equipment Data. – 116,700,000 Btu/hr (34 Mw) capacity; 0.00156 overall fouling factor.

Primary water in tubes; 2,483,000 lbs/hr (~5,000 gpm); inlet temp, 167°F; outlet temp, 120°F; two pass; 5.5 ft/sec flow in tubes; 12 psi pressure drop through tubes.

Secondary water in shell; 3,334,000 lbs/hr (~6,667 gpm); inlet temp, 85°F; outlet temp, 120°F.

Construction. – Design pressure for shell, 150 psi at 200°F; design pressure for tubes, 1000 psi at 200°F; test pressure for shell, 225 psi; test pressure for tubes, 1500 psi.

Tubes: 304L stainless, $^5/_8$ in. OD × 0.035 in. wall; 1190 tubes per bundle, rolled and then welded to tube sheet. Conform to ASTM specification A-249, TP 304L.

Shell: carbon steel except where in contact with primary water; these areas are stainless overlay on carbon steel. Overall dimensions, 33 ft $2^{11}/_{16}$ in. long by 62 in. diam.

Codes: ASME Unfired Pressure Vessel Code, Section VIII, and Nuclear Code Cases 1270 N and 1273 N.

Primary Coolant Pumps 01-PU-1A, -1B, -1C, -1D

Manufacturer

Pumps

Byron-Jackson Company
Los Angeles, California

Manufacturer's Type DVSS 10 × 14 × 20

Motors

Ideal Electric Company
Mansfield, Ohio

Pump Data. – Capacity: 5000 gpm at TDH 365 ft, 1800 rpm; ~700 gpm at 270 rpm. Vertical centrifugal; 304 stainless steel with Graphitar bearings. Design conditions: demineralized water service, 1000 psig at 200°F.

Motor Data. – Main Motor: 600 hp, squirrel cage, 3 phase, 60 cycle, induction motor, vertical shaft. 1800 rpm (130 amps at 2300 v ac); Class B insulation (60°C temperature rise); service factor 1.35.

Pony Motor: 3 hp, series wound, type D; 270 rpm (31 amps at 118 v dc); Class H insulation (60°C temperature rise); service factor 1.35. Shunt field for testing.

Primary Coolant Main Pressurizer Pump 01-PU-4A & 4B

Manufacturer

Allis-Chalmers Manufacturing Company
Milwaukee, Wisconsin

Manufacturer's Size: 4 × 3, type HD

Equipment Data. – 9 stage centrifugal pump, stainless steel. Capacity: 300 gpm; 2400 ft TDH; 0-3485 rpm. Demineralized water service at 200°F.

Drive Motor: Allis-Chalmers Manufacturing Company, Norwood Works, Norwood, Ohio; 300 hp, 3560 rpm, 2300 v, 3 phase, split sleeve bearing, Frame 507US. Type G.

Coupling: Waldron, Waldron-Hartig Corp., New Brunswick, N.J.; $2^1/_2$ AHS spacer type gear coupling.

Variable Speed Drive Unit: Louis Allis Company, Milwaukee, Wisconsin; 375 lb-ft at 3500 rpm. Magnetic clutch type, Frame C-LC 50-20. Assembly No. 1-809300.

Mechanical Seal Assembly: "John Crane," Crane Packing Company, Morton Grove, Illinois. Type 8B1.

Auxiliary Pressurizer Pump 01-PU-11

Manufacturer

Sundstrand-Denver, a Division of Sundstrand
Corporation Industrial Products Group
2480 W. 70th Avenue
Denver, Colorado 80221

Equipment Data. – Vertical centrifugal, single stage, with 3540 to 17,000 rpm speed increaser gearbox.

Capacity: 30 gpm at 1650 TDH demineralized water service at 150°F.

Drive Motor: General Electric; 40 hp, 3540 rpm, 3 phase 440 v, totally enclosed; TEFC Class F 75°C rise; 1.15 service factor.

Secondary Coolant Pumps 03-PU-6A, 6B, and 6C

Manufacturer

Byron Jackson Pumps, Inc.
Los Angeles 54, California

Manufacturer's Size $32RXL_e$

Equipment Data. – Vertical centrifugal, single-stage.

Capacity: 13,000 gpm at 120 ft TDH; 1160 rpm. Cooling tower water service (treated) at 90°F.

Construction: Cast iron construction except for 416 SS impeller shaft and 18-8 SS assembly bolts. Submerged, water-lubricated rubber bearings. Direct coupled drive.

Drive motor: Westinghouse Electric Corp.; 450 hp, 1188 rpm, 3 phase, 60 cycle, 2300 v. Type CSP Lifeline, ball-bearing, vertical motor. Guardistor thermolastic insulation, Frame 681-P dripproof.

Emergency Secondary Coolant Pump 03-PU-14

Manufacturer

Byron Jackson Pumps, Inc.
Los Angeles 54, California

Manufacturer's Size 24KXL

Equipment Data. – Vertical centrifugal type, single-stage.

Capacity: 6000 gpm at 100 ft TDH; 1200 rpm. Cooling tower water service (treated) at 90°F. 3000 gpm at 600 rpm.

Construction: Cast iron and steel construction except for 416 SS impeller shaft and 18-8 SS assembly bolts. Submerged, water-lubricated rubber bearings. Direct coupled drive.

Drive motor: Westinghouse Electric Corp.; 2 speed 150/37.5 hp. 1188/594 rpm, 3 phase, 60 cycle, 440 v. Type 537 CSP Lifeline ball-bearing, vertical motor. Guardistor thermolastic insulation, Frame 581-P. Dripproof.

Pool Coolant Pumps 01-PU-9A & -9B

Manufacturer

Aurora Pump Division
The New York Air Brake Company
Aurora, Illinois

Manufacturer's Size 4R × 5, Type O-J

Equipment Data. – 304 stainless steel; mechanical seal, durametallic Type ROTT inside; bearing, ball type antifriction.

Capacity: 1000 gpm at 210 ft TDH; 3500 rpm.

Service: Demineralized water, 115°F. Hydrostatic test pressure 200 psi, continuous duty.

Drive Motor: Westinghouse, 75 hp, 3500 rpm, 440 v, 60 cycle, 3 phase, open dripproof, Class B encapsulated insulating, Frame 404. US Standard for horizontal motors.

Coupling: Falk Corporation, Steelflex size 8F, Type F.

Main Supply Transformers – Substation Nos. 1, 2, 3, and 4

Manufacturer

General Electric

Equipment Data. – Type OAPT1; Form DM; 60 cycle, 3 phase, 13.8 kv wye, 460 v delta. Continuous rating: 1000 kva, 55°C rise, self-cooled, oil-filled. Serial number: No. 1, 736758 – No. 2, 7731801 – No. 3, 7731581 – No. 4, 7373793.

Main Supply Transformer – Substation No. 5

Manufacturer

Allis-Chalmers

Equipment Data. – Serial No. 2371103. Continuous rating: 5000 kva, self-cooled, 55°C rise; 6667 kva forced air 55°C rise. Class OA/EA; 3 phase, 60 cycle, 13.8 kv delta, 2400 v delta, oil-filled, inert gas cover (N_2).

Instrument Power Supply

Normal. – (No. 6 and No. 7): Standard Transformer Co., Warren, Ohio; Type SWH; 60 cycle, single phase, subtractive polarity. 13.8 kv, 120/240 v. Continuous rating: 50 kva, 55°C rise. Pyranol-filled, self-cooled. Serial Number: No. 6, R333 – No. 7, R3334.

Emergency. – (No. 6A and No. 7A): Marcus Transformer Co., Rahway, N.J.; 60 cycle, single phase, specification 107540; 240/480 v–120/240 v. Continuous rating: 50 kva, self-cooled. Serial Number: No. 6A, 57027 – No. 7A, 57028.

Battery Bank for Failure-Free Systems

Manufacturer

Nife Corporation
Nickel Cadmium Battery Division
Copiague, L.I., N.Y.

General Description. – Manufacturer's Type KB1P-12. Translucent cell container with visible electrolyte level. 1.4 v per cell.

Plates: perforated strips of nickel plated steel with formed pocket to hold active material. Nickel compounds are used for positive plates, cadmium compounds in negative plates.

Electrolyte: Solution of potassium hydroxide in distilled water, specific gravity 1.22. (Specific gravity of electrolyte does not change during charge or discharge)

Charging Unit: Republic Aviation Corp., Farmingdale, L.I., N.Y.; automatic constant potential, battery charger, model RAC 250-50CP(120) SPN.

Diesel Generator

Manufacturer Caterpillar Tractor Company
Peoria, Illinois

Diesel Engine. – Model D379A, 8 cylinder, V-type, $6\frac{1}{4}$ in. bore by 8 in. stroke; 610 hp at 1200 rpm, full load, sea level. Serial No. 68B428, battery start; 68B429, air start.

Generator. – General Electric, Model 5SJ1415A5; 480/277 volt, 3 phase, 60 cycle. Continuous rating: 350 kw, 0.8 power factor, 70°C rise. 526 amp, 438 kva at 1200 rpm; Type AT1, Frame 966; Serial No. TW8350789, battery start; TW8350788, air start.

Special Building Hot Exhaust (SBHE) Fan 05-FN-1, FN-2, and SBHE-3

Manufacturer Westinghouse Electric Corporation
Sturtevant Division
Hyde Park, Boston, Mass.

Manufacturer's Size 8040 ARR 1 SWS1, Airfoil Series 800, Class IV, 40 $\frac{1}{4}$ in. Wheel Diam

Equipment Data. – Capacity: 28,500 cfm at 13 in. water static pressure, 1601 rpm, 70°F.

Construction: Welded steel housing, welded die-formed airfoil blade wheel, ball antifriction bearings, multiple belt drive.

Drive Motor: Westinghouse Electric Corp., Motor and Gearing Division: Buffalo Plant, Buffalo 5, New York. 75 hp, squirrel-cage, Lifeline A; Class B insulation, encapsulated, 60°C rise over 40°C ambient. Tested in accordance with ASA C-50.2 – 1955. (SBHE-3 is identical except is 100 hp.)

Hot Off-Gas Fans – for Open System 05-FN-3 and FN-4

Hot Off-Gas Fans – for Closed System 05-FN-5 and FN-6

Manufacturer Buffalo Forge Company
Buffalo, New York

Manufacturer's Type 35-1 Y "CB" Exhauster (Special) ARR-1

Equipment Data. – Stainless steel, welded construction.

Capacity: 500 cfm, 53 in. water static pressure, 3070 rpm, 7.2 hp, 70°F.

Construction: All-welded, 304 stainless steel housing, with flanged and gasketed inlet side cover, 304L SS wheel; bearings are two deep groove, single row, pillow block. Shaft is fitted with stuffing box and tested to an internal pressure of 60 in. water. Multiple belt drive.

Drive Motor: US Electric Motor Corp., $7\frac{1}{2}$ hp, 3500 rpm, Frame 215, 440 v, 3 phase, 60 cycle. Uniclosed Type H horizontal, "Everseal" encapsulation.

Appendix C

HIGH FLUX ISOTOPE REACTOR REPORTS

ORNL CF Report No. and Date	Title	Author	Description
59-2-65 3/20/59	HFIR Preliminary Design Study	J. A. Lane et al	Description of the HFIR design concept and preliminary design calculations. Superseded by ORNL CF 60-3-33.
59-7-72 7/21/59	Effect of Velocity on Corrosion of Types 1100 and X8001 Aluminum by Distilled Water at 230°C (Loop I: Run 63)	J. L. English	Corrosion rates of proposed HFIR fuel element materials.
59-8-51 8/4/59	Effect of Velocity on Corrosion of Types 1100, 5154-H36, 6061-T6, and X8001 Aluminum by Distilled Water at 260°C (Loop I: Run 64)	J. L. English	Corrosion rates of proposed HFIR fuel element materials.
59-8-109 8/27/59	Status of Plans for Transuranic Element Production	D. E. Ferguson A. Chetham-Strode R. D. Cheverton H. C. Claiborne	A general review of ^{252}Cf production plans. Superseded by ORNL TM-165.
59-8-125 8/31/59	Californium Production in the High Flux Isotope Reactor	H. C. Claiborne M. P. Leitzke	Californium-252 production and heat generation rates in the targets vs cycle time.
59-9-13 9/4/59	Effect of Velocity on Corrosion of Types 1100, 5154-H36, 6061-T6, and X8001 Aluminum by Distilled Water at 290°C (Loop I: Run 65)	J. L. English	Corrosion rates of proposed HFIR fuel element materials.
59-10-17 10/1/59	Effect of Velocity on Corrosion of Types 1100, 5154-H36, 6061-T6, and X-8001 Aluminum by Distilled Water at 200°C (Loop I: Run 66)	J. L. English	Corrosion rates of proposed HFIR fuel element materials.

ORNL CF Report No. and Date	Title	Author	Description
59-10-19 10/8/59	Effect of Different Sets of Cross-Sections on ^{252}Cf Production in the HFIR	H. C. Claiborne	Production rates of ^{252}Cf are calculated using different cross-sections and compared.
59-10-100 10/27/59	A Comparison of Transport Theory and Diffusion Theory Calculations for the High Flux Isotope Reactor	H. C. Claiborne	A comparison of diffusion and transport theories as applied to HFIR. No longer applicable.
59-10-110 10/29/59	HFIR Cooling System Parameter Study	D. W. Vroom	Preliminary evaluation of primary cooling system full power operation requirements for both present and possible future conditions.
59-11-15 11/4/59	Dynamic Corrosion Behavior of Types 1100, 6061-T6, and X8001 Aluminum in Distilled Water at 260°C and 50 psi Mechanical Overpressure (Loop I: Run 67)	J. L. English L. Rice	Corrosion rates of proposed HFIR fuel element materials.
59-11-109 1/11/60	Transplutonium Elements	W. K. Ergen	Possible means of increasing ^{252}Cf production through use of other reactors (ETR) and modifications in HFIR.
59-11-113 11/16/59	Effect of Velocity on Corrosion of Types 1100, X8001F, X2219-T6E46, and 6061-T6 Aluminum by Distilled Water at 260°C and an Average Mechanical Overpressure of 200 psi (Loop I: Run 70)	L. Rice J. L. English	Corrosion rates of proposed HFIR fuel element materials.
59-11-131 11/24/59	Lead Shielding of Spent HFIR Cores	Neil Hilvety	Preliminary shipping container shielding requirements.

ORNL CF Report No. and Date	Title	Author	Description
59-12-16 12/3/59	Effect of Non-Thermal Capture on Californium Production in the HFIR	H. C. Claiborne	Evaluation of the additional ^{252}Cf production which may be gained from fast neutron capture.
59-12-24 12/15/59	Status Report of the Soluble Poison Shim Control for the HFIR	H. A. McLain	Discussion of an early proposal for the HFIR control system. (Not applicable to the present design.)
59-12-50 12/14/59	Effect of Velocity on Corrosion of Types 1100, X8001F, X2219-T6E46, and 6061-T6 Aluminum by Distilled Water at 260°C and an Average Mechanical Overpressure of 140 psi (Loop I: Run 69)	L. Rice J. L. English	Corrosion rates of proposed HFIR fuel element materials.
59-12-89 12/30/59	Effect of Velocity on Corrosion of Types 1100, 6061-T6, and X8001 Aluminum by Distilled Water at 200°C and an Average Mechanical Overpressure of 220 psi (Loop I: Run 71)	L. Rice J. L. English	Corrosion rates of proposed HFIR fuel element materials.
60-2-15 2/1/60	Effect of Velocity on Corrosion of Types 1100, X8001, 5154-H36, and 6061-T6 Aluminum by Distilled Water at 260°C with a Total Pressure of 940 psi (Loop I: Run 72)	L. Rice J. L. English	Corrosion rates of proposed HFIR fuel element materials.
60-2-24 2/5/60	Effect of Surface Preparation and Static Prefilming Treatments on the Corrosion of Type X8001 Aluminum by Flowing Distilled Water at 260°C (Loop I: Run 74)	J. L. English L. Rice	Corrosion rates of proposed HFIR fuel element materials.
60-2-37 2/8/60	Effect of Velocity on Corrosion of Types 1100 and X8001 Aluminum by Distilled Water at 230°C and an Average Mechanical Overpressure of 235 psi (Loop I: Run 73)	L. Rice J. L. English	Corrosion rates of proposed HFIR fuel element materials.

ORNL CF Report No. and Date	Title	Author	Description
60-2-52 2/10/60	Corrosion Behavior of Some Experimental Alloys in Flowing Distilled Water at 260°C (Loop I: Run 75)	J. L. English L. Rice	Corrosion of HFIR fuel element materials.
60-3-10 3/4/60	Effects of Fast Neutron Reactions in the Beryllium Reflector of the HFIR	H. C. Claiborne	Gas formation and physical changes to beryllium as a result of radiation exposure are discussed. Studies are reported on the primary-loop pressure control system and the secondary-loop temperature control system.
60-3-12 3/1/60	Preliminary Hot Spot Analysis of the HFIR	Neil Hilvety	Preliminary HFIR hot spot analysis - method of analysis, variables considered, calculated burnout power levels and hot spot temperatures.
60-3-12 9/18/61 Supplement	Supplement to "Preliminary Hot Spot Analysis of the HFIR"	Neil Hilvety	Report revised to include new correlations for heat transfer coefficient and burnout heat flux.
60-3-33 3/15/60	High Flux Isotope Reactor -- A General Description	T. E. Cole	Preliminary description of the proposed basis for HFIR design.
60-3-40 3/4/60	Some Preliminary Calculations of Shielding, Core Shutdown Cooling Requirements, and Thermal Shield Heat Generation in the HFIR	E. H. Gift	Very preliminary evaluation of water shielding requirements for the reactor and spent cores and afterheat generation and cooling requirements of spent cores. See also ORNL CF 60-4-110.

ORNL CF Report No. and Date	Title	Author	Description
60-4-56 4/11/60	Effect of Velocity on Corrosion of Types X8001, 1100, 5154-H36, and 6061-T6 Aluminum by Distilled Water at 260°C with a Total Pressure of 1030 psi (Loop M: Run 57)	L. Rice J. L. English	Corrosion of HFIR fuel element materials.
60-4-104 4/27/60	Trip Report to the Westinghouse Testing Reactor, April 18, 1960	H. A. McLain	Report of trip to discuss the April 3, 1960 fuel meltdown in the WTR.
60-4-110 4/29/60	Gamma and Beta Heat Generation Rates in the HFIR Core	Neil Hilvety	Heat generation rates and heat fluxes in the fuel element due to fission product heating as a function of time after shutdown.
60-4-120 4/20/60	High Flux Isotope Reactor Critical Experiment - Equipment	C. A. Burchsted	Mechanical facilities description for the first HFIR critical experiment, which used a homogeneous solution as fuel.
60-5-19 5/25/60	Activity Due to ^{16}N and ^{17}N in the HFIR Primary Coolant	H. A. McLain	Calculated results of ^{16}N and ^{17}N activities in the primary coolant during the normal 100-Mw operation of the HFIR.
60-5-33 5/9/60	Comments on WTR Fuel Meltdown	F. T. Binford, W. R. Gambill, and J. F. Wett, Jr.	Authors found that the cause of the WTR fuel elements meltdown of April 3, 1960 probably was due to a nonbonded area in the fuel element. Points out the importance of a good clad bond.
60-5-79 Revised 4/1/61	HFIR Drawing List	J. P. Gill	Obsolete drawing list.

ORNL CF Report No. and Date	Title	Author	Description
60-6-50 6/14/60	Metallographic Examination of Heat Flux Corrosion Test (Type 1100 Aluminum) Specimens 100N-A4 and 100N-A5	T. M. Kegley, Jr.	Corrosion of HFIR fuel element materials.
60-6-52 6/10/60	Activity in the HFIR Primary Coolant System After a Meltdown of the Fuel in Reactor	H. A. McLain	Primary coolant radioactivity after a fuel meltdown within the reactor vessel estimated. Results reported for a reactor containing 6 kg ^{235}U which has operated at 100 Mw for 10 days. (Gaseous fission product activity for a reactor containing 8 kg ^{235}U which has operated at 100 Mw for 15 days is reported in CF 61-6-58.)
60-6-75 6/8/60	Sodium-24 Activity in the HFIR Primary Coolant Water	H. A. McLain	Calculated results of ^{24}Na activity in the primary coolant during normal 100-Mw operation of the HFIR.
60-6-123 6/30/60	Shielding of Pipes in the HFIR Primary Coolant System	H. A. McLain L. A. Haack	Specified shielding for the pipes in the HFIR primary coolant system.
60-7-53 7/25/60	Shielding of Demineralizers and Filters in the HFIR Primary Coolant System	H. A. McLain L. A. Haack	Shielding specified for the HFIR primary coolant demineralizers and filters.
60-8-24 8/2/60	Effect of Velocity on Corrosion of Types 1100, 6061-T6, and X8001-F Aluminum by Distilled Water at 170°C (Loop I: Run 68)	J. L. English L. Rice	Corrosion of HFIR fuel element materials.

ORNL CF Report No. and Date	Title	Author	Description
60-8-39 8/10/60	HFIR Response to Void Swept Into Flux Trap	R. S. Stone	Results of a preliminary analog study of the island void accident. No longer applicable.
60-8-78 8/15/60	Metallographic Examination of Heat Flux Corrosion Test (Type 1100 Aluminum) Specimens 100N-A6 and 100N-A7	T. M. Kegley, Jr.	Corrosion of HFIR fuel element materials.
60-8-87 8/19/60	Summary of Information on HFIR Target Materials and Recommended Work	E. S. Bomar, Jr.	A summary of early fabrication work and recommendations for development work.
60-8-145 8/31/60	HFIR Critical Experiment -2 (HFCE-2)	Paul R. Kasten	Proposed program of experiments to be conducted in the second HFIR critical experiment. Superseded by ORNL CF 61-1-42.
60-9-46 9/12/60	Metallographic Examination of Heat Flux Corrosion (Types 1100 and 6061 Aluminum) Test Specimens 100N-A8, 100N-A9, 100N-A11, and 100N-A12	T. M. Kegley, Jr.	Corrosion of HFIR fuel element materials.
60-10-7 10/3/60	HFIR Reactor Vessel Expansion Problems	W. R. Gall	Preliminary analysis of beam hole deflections due to the thermal expansion and internal pressure in the vessel.
60-11-39 11/14/60	Removal of Radioiodine from Air-Stream Mixtures	R. E. Adams W. E. Browning	Removal of radioiodine from air.
60-12-17 12/2/60	Metallographic Examination of Heat Flux Corrosion (Type 6061 Aluminum) Test Specimens 100N-A13 and 100N-A14	T. M. Kegley, Jr.	Corrosion of HFIR fuel element materials.

ORNL CF Report No. and Date	Title	Author	Description
60-12-18 12/5/60	A Transport Calculation of the HFIR Beam Hole Currents	H. C. Claiborne G. Rakavy	An estimation of beam tube fluxes and energy spectra.
60-12-25 12/6/60	The Corrosion of Types 1100, X8001, and 6061-T6 Aluminum as a Function of Time at Several Flow Rates in Distilled Water at 260°C (Loop M: Runs 58-65)	L. Rice J. L. English	Corrosion of HFIR fuel element materials.
60-12-118 12/29/60	After Shutdown Heating in the HFIR	H. A. McLain	Preliminary after shutdown heat generation rates for the target, the control plates, and the beryllium reflector.
61-1-23 1/10/61	Recommended Hydraulic Measurements in the HFIR Mockup	R. J. Kedl	Preliminary evaluation of information which might be obtained from the HFIR hydraulic mockup.
61-1-31 1/13/61	HFIR Target Material Behavior Metallography Report No. 368	E. S. Bomar D. M. Hewette II	Preliminary information on the probable compatibility of oxides of plutonium and/or americium and curium with aluminum, and dimensional stabilities of the respective mixtures.
61-1-42 1/16/61	Revised Version of HFIR Critical Experiment-2 (HFCE-2)	P. R. Kasten R. D. Cheverton	Experiment to be conducted in the second HFIR critical experiment.
61-1-82 1/25/61	Multiple Controllers for a Single Process	C. H. Weaver	Discussion of concepts concerned with the application of multiple controllers to a single process having one outlet to be controlled.

ORNL CF Report No. and Date	Title	Author	Description
61-2-24 2/6/61	Metallographic Examination of Heat Flux Corrosion (Type 6061 Aluminum) Test Specimens 100N-A16 and 100N-A17	T. M. Kegley, Jr.	Corrosion of HFIR fuel element materials.
61-2-26 2/13/61	Probability That the Spacing Between Two Adjacent Fuel Plates in the HFIR Fuel Element Will Meet Acceptance Standards	D. A. Gardiner	Calculations to determine the probability that the spacing between two adjacent fuel plates in the inner annulus fuel element will meet acceptance standards.
61-2-36 2/2/61	Fuel-Cycle Analysis and Proposed Fuel and Burnable Poison Distribution and Loading for the HFIR and HFCE-2	R. D. Cheverton	Fuel burnup and required radial fuel distribution calculation.
61-2-81 2/21/61	HFIR Beryllium Reflector - Preliminary Design Report	Neil Hilvety	Beryllium reflector design criteria, design calculations, and description. Preliminary.
61-3-82 3/16/61	HFIR Pool Criteria	Neil Hilvety L. A. Haack J. R. McWherter	Design criteria and expected performance and data for pool components.
61-3-82 Addendum No. 1 7/18/61	Addendum to HFIR Pool Criteria Report CF 61-3-82	Neil Hilvety L. A. Haack J. R. McWherter	Design criteria and expected performance and data for pool components.
61-4-55 4/18/61	HFIR Permanent Reflector Flow Test	D. T. Jones W. H. Kelley, Jr.	Experimental evaluation of pressure drop requirements across permanent reflector.
61-4-73 4/18/61	Some Temperature Aspects of the HFIR Control Plate System	L. A. Haack	Preliminary thermal analysis of the HFIR control plates.

ORNL CF Report No. and Date	Title	Author	Description
61-5-33 5/2/61	HFIR Control Rod Seal Test - Interim Report	D. T. Jones W. H. Kelley, Jr.	Test results on high pressure HFIR control rod seals.
61-5-89 5/19/61	Metallographic Examination of Heat Flux Corrosion (Type 6061 Aluminum) Test Specimens 100N-A18 and 100N-A19	T. M. Kegley, Jr.	Corrosion of HFIR fuel element materials.
61-6-24 6/7/61	Comments and Questions Concerning the HFIR Design	The HFIR Design Review Committee	A listing of questions resulting from preliminary evaluation of the HFIR design.
61-6-58 6/16/61	Shielding of the HFIR Primary Coolant Deaerator and Condensers	H. A. McLain	Shielding specified for the HFIR deaerator and condensers.
61-7-50 7/20/61	The ^{235}U Content of Foils Punched from HFIR Critical Experiment No. 2 Fuel Plates	D. W. Magnuson	Evaluation of critical experiment fuel element fabrication results.
61-7-60 7/20/61	After Shutdown Cooling Requirements in the HFIR	Neil Hilvety	Forced circulation cooling requirements for fuel element after shutdown.
61-7-87 7/21/61	Xenon Chase and Samarium Burnup in the HFIR	R. D. Cheverton	Requirements and ability to restart a scrammed HFIR core.
61-8-37 8/15/61	Solution Critical Experiments for the High Flux Isotope Reactor: Preliminary Calculations	H. C. Claiborne	Comparison of theoretical and experimental results.
61-8-52 8/15/61	Preliminary Evaluation of Measurements on HFIR Critical Assembly	J. W. Tackett	Evaluation of fuel element fabrication for second HFIR experiment.

ORNL Report No. and Date	Title	Author	Description
61-8-58 8/18/61	Preliminary Analysis of the HFIR Control Rods	T. G. Chapman	Preliminary analysis of hydraulic loads on HFIR control plates. In general, not applicable to present design.
61-8-62 8/21/61	Natural Convection Cooler for Defective Fuel Elements from the HFIR - Design Calculations	H. C. Claiborne	Design of the defective element, reactor pool heat exchanger.
61-8-75 8/25/61	Preliminary Design of the HFIR Core and Pressure Vessel Assembly	J. R. McWherter	Description of the design of the HFIR core and pressure vessel assembly. (In 1962, to permit distribution to DTIE, number ORNL TM-172 was assigned to this report.)
61-9-52 9/27/61	High Flux Isotope Reactor Critical Experiment No. 2	D. W. Magnuson	Results of the second HFIR critical experiment (Part I).
61-10-65 10/18/61	HFIR Process Control System Studies	S. J. Ball	Studies are reported on the primary loop pressure control system and the secondary-loop temperature control system.
61-10-70 10/31/61	Preliminary Report on Experimental Power and Flux Distribution in HFCE-2	R. D. Cheverton	Power and flux distributions in HFIR critical experiment No. 2
61-11-32 11/20/61	HFIR Control Rod Drive Design Report	C. W. Collins	Description of preliminary design of HFIR control rod drives. See CF 67-11-34 for final design.
62-1-11 1/3/62	Revised HFIR Calculations for the Production of ^{149}Pm from the 147 Chain	R. D. Cheverton	A part of the HFIR fission products poisoning calculations.

ORNL CF Report No. and Date	Title	Author	Description
62-1-31 1/16/62	Design of the TRU-HFIR Target	W. E. Unger J. P. Nichols W. C. Thurber	Target design basis for TRU facility.
62-1-34 1/16/62	Metallographic Examination of Heat Flux Corrosion (Type 6061 Aluminum) Test Specimens 100-0_a-A28 and 100N-A29, and (Type X8001 Aluminum) Test Specimen 100-0_a-A30	T. M. Kegley, Jr.	Corrosion of HFIR fuel element materials.
62-1-52 1/30/62	Summary of HFIR Hot Spot Studies	Neil Hilvety T. G. Chapman	HFIR hot spot analysis - method of analysis, variables considered, calculated burnout power levels and hot spot temperatures.
62-1-53 1/16/62	High Flux Isotope Reactor Critical Experiment No. 2. Part III. Reactivity Measurements in HFIRCE No. 2	D. W. Magnuson	Results of the second HFIR critical experiment (Parts II, III, IV, and V).
62-1-63 1/15/62	High Flux Isotope Reactor Critical Experiment No. 2. Part II.	D. W. Magnuson	Results of the second HFIR critical experiment (Parts II, III, IV, and V).
62-5-10 5/8/62	High Flux Isotope Reactor Critical Experiment No. 2. Part V. Reactivity Measurements with Modified Control Plates	D. W. Magnuson	Results of the second HFIR critical experiment (Parts II, III, IV, and V).
62-5-20 5/2/62	High Flux Isotope Reactor Critical Experiment No. 2. Part IV. Outer Control Plate Sensitivity Values	D. W. Magnuson	Results of the second HFIR critical experiment (Parts II, III, IV, and V).

ORNL CF Report No. and Date	Title	Author	Description
62-5-31 5/8/62	Heat Evolution in HFIR First Cycle Target Rods	E. D. Arnold	Heat generation rates in target vs time in cycle. See also ORNL CF 62-7-34.
62-6-14 6/4/62	Transuranium Processing Development News Letter for May 1962.	D. E. Ferguson	Status of HFIR target work.
62-7-8 7/2/62	Transuranium Processing Development News Letter for June 1962	D. E. Ferguson	Status of HFIR target work.
62-7-21 7/10/62	The Thermal Conductivity at 75°C of Cold-Pressed Al-Gd_2O_3 Pellets	D. L. McElroy W. Fulkerson T. Kollie	Evaluation of thermal conductivity of a pellet which may be similar to those in the HFIR target.
62-7-30 7/16/62	Boiling-Burnout Research at the Savannah River Laboratory	W. R. Gambill	Preliminary report on the effect of hot spot size and orientation on burnout.
62-7-34 7/11/62	Heat Evolution in HFIR First Cycle Target Rods: Second Evaluation	E. D. Arnold	Heat generation rates in target vs time in cycle.
62-7-85 7/25/62	Thermal Expansion Characteristics of HFIR Target Pellets	D. M. Hewette II	Thermal expansion of prototype HFIR target pellets with 11 vol % Gd_2O_3 as a stand-in was determined in the range of 25 to 400°C.
62-8-6 8/1/62	Transuranium Processing Development News Letter for July 1962	D. E. Ferguson	Status of HFIR target work.

ORNL CF Report No. and Date	Title	Author	Description
62-8-25 8/2/62	High Flux Isotope Reactor No. 2. Part VI. Additional Control Plate and Safety Calibrations	D. W. Magnuson	The early HFIR design (now obsolete) control rods with a 20-in. gray region were calibrated under various conditions.
62-9-53 9/24/62	Remarks on Thermal Buckling of Columns, Plates and Shells with Initial Sinusoidal Distortion (HFIR Fuel Plates)	R. N. Lyon	An approximate prediction is given for lateral displacement of an HFIR fuel plate as a function of the temperature difference between the fueled portion and the side plates.
62-9-75 9/13/62	High Flux Isotope Reactor No. 2. Part VII. Reactivity and Power Distribution Measurements with Two-Section Control Plates	D. W. Magnuson	Power distributions in the fuel region were measured and control rods were calibrated, using the early HFIR design (now obsolete) with a 20-in. gray region.
62-10-76 10/24/62	List of HFIR Reports as of September 1962	J. R. McWherter	HFIR reports listed.
62-10-76 Supp. No. 1 7/25/63	Supplementary List of HFIR Reports as of May 31, 1963	J. R. McWherter	Report list updated.
62-10-105 10/31/62	High Flux Isotope Reactor Critical Experiment No. 2. Part VIII. Reactivity and Power Distribution Measurements with 4-3/4-in.-long Middle Sections in the Control Plates	D. W. Magnuson	Power distributions in the fuel region and subcritical reactivities for various insertions of control rods with 4-3/4-in. gray regions were measured.

ORNL CF Report No. and Date	Title	Author	Description
62-11-50 11/20/62	Fission Yield Curves for ^{241}Pu, ^{241}Am, ^{242}Am, and ^{245}Cm	E. D. Arnold	Fission yield curves for ^{241}Pu, ^{241}Am, ^{242}Am, and ^{245}Cm have been obtained by extrapolation of functions of known data for the more common fissionable isotopes.
63-2-5 2/4/63	Analytical Results and Macroscopic Thermal Neutron Absorption Cross-Section Values for Impurities in Type 6061 Al	R. J. Beaver	The concentration of impurities in Type 6061 Al were evaluated from the standpoint of analytical determination.
63-2-52 2/21/63	Gamma Ray and Neutron Heat Generation in the HFIR	D. R. Vondy	The heat generation in the HFIR is given as a function of position and material for both gamma and neutron radiation.
63-4-11 4/5/63	Distribution and Loading of Fuel and Burnable Poison for HFIRCE-3	R. D. Cheverton	Radial fuel and burnable poison distributions are specified for both the inner and outer HFIR fuel annuli.
63-5-45 5/9/63	Preliminary Analysis of HFIR Transients Resulting from Ramp Reactivity Additions	N. Hilvety R. D. Cheverton O. W. Burke	The transient behavior of the HFIR when subjected to ramp reactivity additions was investigated to determine the safety system design requirements for preventing core damage.
63-8-67 8/27/63	Calculations for the Irradiation of Four TRU-HFIR Target Protytypes in the MTR on the ORNL-43 Program	S. D. Clinton	Calculations for irradiation of target prototypes in the MTR.

ORNL CF Report No. and Date	Title	Author	Description
63-9-29 9/12/63	Trip Report on Visit to NRTS, August 27 and 28, 1963	J. C. Griess G. H. Jenks J. R. McWherter	Observations concerning the water chemistry and corrosion in the MTR and ETR, which have water cooling systems similar to that of the HFIR.
63-10-64 10/28/63	Gamma and Neutron Radiation from As-Fabricated TRU-HFIR Target Rods	J. P. Nichols	Calculations of the gamma and neutron dose rates from as-fabricated HFIR target rods are presented.
63-12-38 12/17/63	HFIR Critical Experiment No. 3 - (HFIRCE-3)	R. D. Cheverton	Primary purpose of experiment to check power distribution with a heavier fuel loading, to calibrate the new control rods, and to determine shutdown margins.
64-1-65 1/24/64	Adhesives for Bonding Polyethylene to Type 6061 Aluminum	M. M. Martin	Adhesives for bonding polyethylene to Type 6061 aluminum.
64-1-72 1/31/64	HFIR Review Committee - Meeting No. 1	J. A. Lane	Re-evaluation of the technical feasibility, utility, and safety of the reactor design. Review Committee questions and Project answers concerning purpose, status, and cost.
64-2-5 2/5/64	HFIR Review Committee - Meeting No. 2	J. A. Lane	Review Committee questions and Project answers concerning the nuclear design considerations.
64-2-6 2/6/64	HFIR Review Committee - Meeting No. 3	J. A. Lane	Review Committee questions and Project answers concerning the fuel element and control region design considerations.

ORNL CF Report No. and Date	Title	Author	Description
64-2-19 2/11/64	HFIR Review Committee - Meeting No. 4	J. A. Lane	Review Committee questions and Project answers concerning the fuel element design and fabrication.
64-2-20 2/12/64	HFIR Review Committee - Meeting No. 5	J. A. Lane	Review Committee questions and Project answers concerning the primary cooling system components and shielding.
64-2-53 2/26/64	HFIR Review Committee - Meeting No. 6	J. A. Lane	Review Committee questions and Project answers concerning the core structure and core hydraulics.
64-2-73 2/3/64	Thermal Deflection Test - HFIR Outer Control Plate	G. J. Dixon	Description of test conducted in late 1962 to determine the thermal deflection encountered in a simulated HFIR outer control plate.
64-3-12 3/4/64	HFIR Review Committee - Meeting No. 7	J. A. Lane	Review Committee questions and Project answers concerning the beryllium reflector design and the reactor pool system.
64-4-44 4/13/64	HFIR Review Committee - Meeting No. 8	J. A. Lane	Review Committee questions and Project answers concerning the control region design and control rod fabrication.
64-4-68 4/14/64	HFIR Review Committee - Meeting No. 9	J. A. Lane	Review Committee questions and Project answers concerning the reactor control and instrumentation.

ORNL CF Report No. and Date	Title	Author	Description
64-4-69 4/15/64	HFIR Review Committee - Meeting No. 10	J. A. Lane	Review Committee questions and Project answers concerning the off-gas system and hot waste disposal.
64-4-70 4/16/64	HFIR Review Committee - Meeting No. 11	J. A. Lane	Review Committee questions and Project answers concerning the target region design and fabrication.
64-4-71 4/17/64	HFIR Review Committee - Meeting No. 12	J. A. Lane	Review Committee questions and Project answers concerning the reactor operation procedure.
64-4-72 4/17/64	HFIR Review Committee - Meeting No. 13	J. A. Lane	Review Committee questions and Project answers concerning the reactor safety and containment.
64-5-39 5/11/64	Prediction of Critical Heat Flux for Natural Convection of Water in Blocked Vertical Channels	W. R. Gambill	Predictions of the minimum critical heat flux for natural convection of water in a heated vertical channel closed at the bottom and open to a liquid supply at the top.
64-5-59 5/18/64	Analysis of HFIR Irradiated Fuel Element Shipping Cask	J. H. Evans G. H. Llewellyn T. W. Pickel A. E. Spaller B. W. Wieland	Report presents the pertinent shielding, heat transfer, and stress calculations required and compares the analytical results with specific allowable values as established in AEC regulations. This report reissued as ORNL-TM-959.

ORNL CF Report No. and Date	Title	Author	Description
64-7-74 7/27/64	High Flux Isotope Reactor Critical Experiment No. 3. Part I. The ^{235}U Content of Foils Punched from Fuel Plates	J. T. Thomas	Evaluation of critical experiment fuel element fabrication results.
64-9-36 9/18/64	Reactor Operations Review Committee Review of "The High Flux Isotope Reactor - Accident Analysis" (Second Draft, July 1964)	Reactor Operations Review Committee	Report presents the conclusions and recommendations of RORC following review of the second draft of the hazards report.
64-10-8 10/5/64	The Curium-Americium Slug Cask Transfer	J. P. Nichols and F. L. Peishel	Description of a transfer cask for slugs containing transplutonium elements such as a 5-month irradiated HFIR target. Heat transfer problems during shipment are discussed.
64-11-7 11/3/64	HFIR Review Committee Meeting - October 14, 1964	J. A. Lane	Overall review meeting to consider changes in design and new developments in technology evolving during the period of the review.
64-12-43 12/18/64	Preliminary HFIR Hydraulic Tests	G. J. Dixon	Data collected during test period (9/25/64 to 11/26/64) are presented.
65-2-64 2/26/65	High Flux Isotope Reactor Critical Experiment No. 3. Part II. Results of Antimony-124 -Source Experiments	J. T. Thomas	Evaluation of critical experiment antimony source fabrication results.
65-4-21 4/12/65	HFIR Review Committee Meeting - December 16, 1964	J. A. Lane	Overall review meeting to consider changes in design and new developments in technology evolving during the period of the review.

ORNL CF Report No. and Date	Title	Author	Description
65-7-31 7/9/65	Calculated Waiting Time for Removal of a Spent HFIR Fuel Element	L. A. Haack T. G. Chapman	Waiting times for removal of a 15-day, 100-Mw fuel element, cooled by natural convection, were calculated for uniform coolant gaps in the range from 30 to 50 mils.
65-7-48 7/19/65	Prevention of Release of Radioiodine in Nonelemental Forms at ORNL	W. E. Browning	Prevention of release of radio-iodine in nonelemental forms at ORNL.
65-7-60 7/29/65	HFIR Review Committee Report	J. A. Lane	Overall review meeting to consider changes in design and new developments in technology evolving during the period of the review. Individual comments of each committee member on the reactor design are appended.
65-7-66 7/21/65	Hydraulic Tests of the HFIR Target Region	R. J. Kedl	Results of hydraulic tests performed on the target region components and assemblies are presented.
65-8-2 8/2/65	Power Deposition in the HFIR Moderator from Fast Neutron Scattering Collisions	T. M. Sims R. D. Cheverton	Power deposition in the HFIR moderator from fast neutron scattering collisions.
65-8-12 8/5/65	Reactor Operations Review Committee Preoperational Review of HFIR	Reactor Operations Review Committee	Results of review presented and startup recommended.
65-8-55 4/1/66	Experiments for the Development of a Reactivity Acceptance Test for HFIR Fuel Elements	J. T. Thomas S. J. Raffety	Evaluation of experiments performed to develop and calibrate a reactivity acceptance test for HFIR fuel elements.

ORNL CF Report No. and Date	Title	Author	Description
65-8-57 8/24/65	Critical Experiments with a HFIR Fuel Element	S. J. Raffety J. T. Thomas R. K. Reedy, Jr.	Evaluation of experiments performed to determine the reactivity of the first fuel element fabricated for use in the HFIR.
65-9-7 9/30/65	Revised HFIR Shutdown Heat Generation Rates	T. G. Chapman	Shutdown heat generation rates for a HFIR core loading revised to reflect dimensional changes, fabrication tolerances, and operational uncertainties.
65-9-59 9/17/65	Preliminary "Turnaround" Tests of a Simulated HFIR Target Unit	L. A. Haack T. G. Chapman	Preliminary tests indicate that a target rod can be safely removed as soon as 20 minutes after shutdown.
65-11-1 11/30/65	HFIR Alignment Manual	W. H. Kelley, Jr.	Collection of procedures, data, and general information on the alignment, indexing, and fastening of the structural components.
65-11-29 11/12/65	High Flux Isotope Reactor - Safety Review Questions and Answers	T. E. Cole	Questions raised by AEC concerning the HFIR Safety Analysis, and answers by ORNL, are detailed in this report. Also included in ORNL-3573.
65-11-29 Supp. No. 1 2/1/66	High Flux Isotope Reactor - Safety Review Questions and Answers	T. E. Cole	Additional questions raised by AEC concerning the HFIR Safety Analysis, and answers by ORNL, are detailed in this report. Also included in ORNL-3573.

ORNL CF Report No. and Date	Title	Author	Description
65-11-29 Supp. No. 2 8/15/66	High Flux Isotope Reactor - Safety Review Questions and Answers	T. E. Cole	Additional questions raised by AEC concerning the HFIR Safety Analysis, and answers by ORNL, are detailed in this report. Also included in ORNL-3573.
65-12-2 12/1/65	Analysis of Selected HFIR Critical Experiment Data for Period August 25, through October 22, 1965	R. D. Cheverton T. M. Sims	Analysis of experiments performed to verify accuracy of nuclear characteristics of core.
66-1-71 1/28/66	RERC Review of the Joint ANL-ORNL Beam Spectrum Measurements at the HFIR	R. C. Weir	RERC review of proposed joint ANL-ORNL beam spectrum measurements at the HFIR.
66-2-10 2/8/66	The Reactivity of the First Commercially Manufactured HFIR Fuel Element	E. B. Johnson	Report of test to determine reactivity of first commercially manufactured HFIR fuel element.
66-3-68 3/15/66	Data Report - Postirradiation Examination of HFIR Target Rods TRU-2 and TRU-3	D. E. Wilson	Experiment conducted to determine the adequacy of heat transfer across the pellet-to-cladding cap, the effect of cyclic thermal strain across the fins, the transformation of X-8001 aluminum to silicon, pressure buildup from fission gases, and swelling from solid fission products.
66-4-4 4/4/66	The Reactivity of a Commercially Manufactured HFIR Fuel Element: Memo No. 2	E. B. Johnson	Report of reactivity acceptance test on fuel element consisting of 2-0 and 3-I.
66-4-10 4/5/66	HFIR Critical Experiment No. 3, Part III	J. T. Thomas	Evaluation of critical experiment fuel fabrication results.

ORNL CF Report No. and Date	Title	Author	Description
66-5-38 5/12/66	1965 RORC Review of the High Flux Isotope Reactor (HFIR)	Reactor Operations Review Committee	Annual review of HFIR by RORC (reactor operating at 20 Mw).
66-6-62 6/21/66	HFIR Quarterly Report - January, February, and March of 1966	R. V. McCord B. L. Corbett	Operating report for first quarter of 1966.
66-7-11 7/7/66	HFIR Neutron Fluctuation Spectra During Increase of Power to 50 Mw	D. N. Fry	Neutron fluctuation measurements were made to detect any unusual resonances in the frequency spectrum of the neutron density fluctuations.
66-7-27 7/15/66	The Reactivities of Several HFIR Fuel Elements: Memo No. 3	E. B. Johnson	Report of reactivity acceptance tests on seven fuel elements: 3-O, 2-I; 4-O, 4-I; 5-O, 5-I; 6-O, 6-I; 7-O, 7-I; 8-O, 8-I; 9-O, 9-I.
66-8-31 8/16/66	HFIR Critical Experiment No. 3. Part IV. "Eu_2O_3-Ta-Al Control Cylinder Calibration and Experiments Supporting HFIR Startup"	J. T. Thomas	Evaluation of critical experiments to evaluate reactivity control and power distribution under several conditions.
66-8-48 8/3/66	Data Report - Postirradiation Examination of HFIR Target Rod TRU-4	D. E. Wilson G. A. Moore	Experiment performed to determine adequacy of heat transfer across the pellet-to-cladding gap, the effect of cyclic thermal strain across the fins, the transformation of X-8001 aluminum to silicon, the pressure buildup from fission gases, and the swelling from solid fission products.

ORNL CF Report No. and Date	Title	Author	Description
66-8-64 8/26/66	HFIR Operating Report for the Period Through December of 1965	R. V. McCord B. L. Corbett	Operating report for the period through December 1965. (First issue of periodic operating reports on the HFIR.)
66-9-16 9/12/66	The Reactivities of Two High Flux Isotope Reactor Fuel Elements, Nos. 10 and 11: Memo No. 4	E. B. Johnson	Report of reactivity acceptance tests on two fuel elements: 10-O, 10-I; 11-O, 11-I.
66-10-29 10/14/66	Information Pertaining to HFIR Fuel Plate Radiation Damage Studies	R. D. Cheverton T. M. Sims	Evaluation of radiation damage in terms of fuel-plate mechanical integrity and heat-removal capabilities.
66-10-74 10/28/66	High Flux Isotope Reactor Quarterly Report - April, May, and June of 1966	R. V. McCord B. L. Corbett	Operating report for the second quarter of 1966.
67-1-55 1/11/67	Spatial and Energy Distributions of the Fast Neutron Flux: A Comparison of Calculations with Measurements in the HFIR	W. A. Hartman	Evaluation of calculations of space and energy distribution of the neutron flux.
67-5-36 5/17/67	The Reactor Experiment Review Committee's Review of the HFIR Central Hydraulic Tube Irradiation Facility	R. C. Weir	Summary of RERC review meetings on the installation and operation of a hydraulic tube irradiation facility in the HFIR target region.
67-6-2 6/2/67	1966 RORC Review of the High Flux Isotope Reactor (HFIR)	Reactor Operations Review Committee	Annual review of HFIR by RORC.

ORNL CF Report No. and Date	Title	Author	Description
67-6-4 6/5/67	The Reactivities of Several High Flux Isotope Reactor Fuel Elements: Memo No. 5	E. B. Johnson R. K. Reedy	Report of reactivity acceptance tests on 13 fuel elements: 12-O, 12-I; 13-O, 13-I; 14-O, 14-I; 15-O, 15-I; 16-O, 16-I; 17-O, 17-I; 18-O, 18-I; 19-O, 19-I; 20-O, 20-I; 21-O, 21-I; 22-O, 22-I; 23-O, 23-I; 24-O, 24-I.
67-6-52 7/10/67	The Generation of Hydrogen in TRU-HFIR Targets	J. P. Nichols	Estimates of hydrogen generation rates in TRU-HFIR targets during irradiation in the HFIR island.
67-8-2 8/2/67	Reactor Operations Review Committee Meeting July 25, 1967, HFIR Control Plate Mechanical Problems	Reactor Operations Review Committee	Discussion of control plate bearing bracket failure problem.
67-8-30 8/18/67	Reactor Operations Review Committee Meeting August 17, 1967, HFIR Control Plate Bearing Support Problem	Reactor Operations Review Committee	Report presents recommendations of RORC on the control plate bearing support problem.
67-8-54 8/29/67	Reactor Operations Review Committee Meeting Held August 29, 1967, on the HFIR Control Plate Bearing Problem	Reactor Operations Review Committee	Modifications to bearings and tracks were reviewed and startup recommended.
67-10-1 10/2/67	Reactor Operations Review Committee Meeting 9/29/67, HFIR Control Plate Mechanical Problems	Reactor Operations Review Committee	Further discussion of control plate bearing mounting lug failures. Recommendation made to delay startup until both the mechanism of failure and its cure are more convincingly demonstrated, and a test program to assure safe operation is established.

ORNL CF Report No. and Date	Title	Author	Description
67-11-6 11/1/67	The Reactivities of High Flux Isotope Reactor Fuel Elements 25 Through 41: Memo No. 6	E. B. Johnson R. K. Reedy	Report of reactivity acceptance tests on 17 additional fuel elements - 25 through 41.
67-11-20 11/20/67	List of Reports Pertaining to the HFIR Project	HFIR Staff	HFIR reports listed.
67-11-34 11/10/67	Control Element Drives for the High Flux Isotope Reactor	J. E. Jones, Jr.	Description of the HFIR rod drive system including gear boxes for the shim safety plates and shim regulating cylinder. The safety release mechanism is included in this description.
67-11-35 11/17/67	Reactor Operations Review Committee Meetings Held November 15, 16, 17, 1967, on the HFIR Control Plate Bearing Problems	Reactor Operations Review Committee	RORC recommendations on control plate bearing problems.
67-11-50 11/22/67	Ad Interim Review Panel for HFIR Control Plates; Progress Report No. 1	A. A. Abbatiello A. P. Fraas W. R. Martin T. L. Trent, Jr.	Review Panel recommendations on control plate bearing problems.

ORNL Report No. and Date	Title	Author	Description
ORNL-TM-99 12/12/61	Cooling of the HFIR Beryllium Reflector Following a Reactor Scram or an Electrical Power Outage	H. A. McLain	Thermal stresses in the beryllium reflector were calculated for conditions following a scram with simultaneous loss of coolant flow and for an electrical power outage.
ORNL-TM-114 11/15/61	Void Coefficient of Reactivity Associated with the Island Region of the High Flux Isotope Reactor	R. D. Cheverton	Examination of the HFIR void accident.
ORNL-TM-165 5/18/62	Transuranium Element Research and Production - ORNL Contribution to FY 1962 Research Supplement to the AEC Annual Report to Congress	D. E. Ferguson A. Chetham-Strode J. R. McWherter	Brief description of the research, production, and chemical processing of transuranium elements.
ORNL-TM-172 8/25/61	Preliminary Design of the HFIR Core and Pressure Vessel Assembly .	J. R. McWherter	Description of the design of the HFIR core and pressure vessel assembly. (This report was originally issued as CF 61-8-75. TM number assigned to permit distribution to DTIE.)
ORNL-TM-274 8/17/62	The Detection of Boiling in a Water-Cooled Nuclear Reactor	A. L. Colomb F. T. Binford	A description of a method to detect the onset of boiling by the analysis of acoustical and nuclear noise.
ORNL-TM-299 7/27/62	An Estimate of the Effect of Neutron Energy Spectrum on Radiation Damage of Steel	H. C. Claiborne	Effect of the neutron energy spectrum on the radiation damage of the pressure vessel.

ORNL Report No. and Date	Title	Author	Description
ORNL-TM-456 4/1/63	Pressure Equalization by Fluid Exchange Between Parallel Flow Channels	T. G. Chapman P. N. Stevens	Pressure equalization by cross-flow of water between parallel channels through holes was evaluated experimentally for use in the HFIR control rod region hydraulic studies.
ORNL-TM-647 10/1/63	Compatibility of Cadmium-Bearing Materials with the HFIR Fuel Plate Constituents	T. D. Watts K. K. Sinha	Compatibility study between various cadmium-bearing materials and the filler section constituents, aluminum and B_4C.
ORNL-TM-661 2/17/64	Thermal-Fatigue Analysis of the HFIR	J. T. Venard A. E. Carden	Superposed deformations from analyses of simple elements led to recommendation of relieving thermal strain by removing axial restraints.
ORNL-TM-778 10/28/63	High Pressure Test of the HFIR Reciprocating Rod Seal	W. H. Kelley, Jr. R. M. Hill, Jr.	Evaluation of two control rod seal housing designs and several materials combinations of drive rods, seal rings, and seal housings for use in the HFIR.
ORNL-TM-811 6/64	Fabrication and Preirradiation Data for HFIR Prototype Target Rods	J. D. Sease D. M. Hewette II	Fabrication and preirradiation data for HFIR prototype target rods for irradiation in ETR.
ORNL-TM-849 4/27/64	The Effect of Gamma Radiation on the Thermal Conductivity of Aluminum Corrosion Product Films	J. C. Griess H. C. Savage J. M. Baker	Experiment to determine the effect of gamma radiation on the thermal conductivity of corrosion product films (boehmite) formed on aluminum surfaces in water.

ORNL Report No. and Date	Title	Author	Description
ORNL-TM-902 8/64	Specifications for HFIR Fuel Elements HFIR-FE-1	G. M. Adamson, Jr. J. R. McWherter	Specification covers requirements for the two types of aluminum-base fuel elements which constitute the HFIR fuel assembly. (Specification is currently being revised.)
ORNL-TM-959 1/65	Analysis of HFIR Irradiated Fuel Element Shipping Cask	J. H. Evans G. H. Llewellyn T. W. Pickel A. E. Spaller B. W. Wieland	Analysis of shielded transfer cask to establish compliance with requirements for Bureau of Explosives approval and AEC regulations.
ORNL-TM-1029 6/65	Laboratory Corrosion Studies for the High Flux Isotope Reactor	J. L. English J. C. Griess	Results of static tests conducted as the first phase of the corrosion test program.
ORNL-TM-1030 9/66	Dynamic Corrosion Studies for the High Flux Isotope Reactor	J. L. English J. C. Griess	Results of dynamic tests carried out in a stainless steel pump loop as the second phase of HFIR corrosion test program.
ORNL-TM-1084 9/3/65	HFIR Target Design Study	T. G. Chapman	Studied effects of ^{242}Pu loading, number of rods, neutron flux, cycle time, and pellet-tube conductance to recommend initial target rod loading.
ORNL-TM-1095 6/65	Fabrication and Preirradiation Data of HFIR Target Elements for Savannah River Irradiations	J. D. Sease	Description of design and fabrication of target elements containing ^{242}Pu fabricated for irradiation at Savannah River, together with results of examination of completed elements.

ORNL Report No. and Date	Title	Author	Description
ORNL-TM-1138 6/65	Operating Manual for the HFIR - Description of the Facility	HFIR Staff	Description of the HFIR facility.
ORNL-TM-1168 7/65	Operating Manual for the HFIR - Operating Procedures	HFIR Staff	Operating procedures for the HFIR.
ORNL-TM-1193 9/65	Calculations of Charge and Contour Dimensions of Powder-Loading Assembly Used in the Production of HFIR Composite Fuel Compacts	T. D. Watts W. J. Werner J. P. Hammond	Presents method for calculating the dimensions of a powder-loading assembly and attendant fuel and filler compositions (as applied to inner-annulus plates) required for the production of the precision composite compact to the stringent loading and contour specifications.
ORNL-TM-1259 9/1/65	A Second Generation of Reactor Control Systems as Applied to the HFIR	L. C. Oakes	A safety system, a start-up instrument covering 10 decades, and a multiple channel regulating system--all transistorized and employing modular construction--are described as applied to the HFIR.
ORNL-TM-1291 10/1/65	The Release and Adsorption of Methyl Iodide in the HFIR Maximum Credible Accident	R. E. Adams W. E. Browning, Jr. Wm. B. Cottrell G. W. Parker	Study undertaken to establish both the maximum percentage of the iodine inventory which could exist in the containment vessel as methyl iodide, and the efficiency of the HFIR-SBHE system filter for methyl iodide.

ORNL Report No. and Date	Title	Author	Description
ORNL-TM-1332 12/29/65	Neutron Density Fluctuations Induced by Hydraulic Noise in a Nuclear Power Reactor	B. R. Lawrence	Feasibility study to determine whether or not reactor parameters can be obtained by suitable analysis of the random fluctuations of certain physical variables in the power reactor system.
ORNL-TM-1348 1/66	Influence of Irradiation on the Tensile Properties of the Aluminum Alloy 6061	H. E. McCoy, Jr. J. R. Weir	Study made to determine whether the mechanical properties of the aluminum alloy 6061 are altered significantly by irradiation.
ORNL-TM-1372 1/66	HFIR Pressure Vessel and Structural Components Material Surveillance Program	J. R. McWherter R. E. Schappel J. R. McGuffey	Reviews design and fabrication precautions against brittle fracture, and outlines materials- and corrosion-surveillance program.
ORNL-TM-1377 2/66	Fabrication of Aluminum-Base Irradiation Test Plates	M. M. Martin W. J. Werner C. F. Leitten, Jr.	Describes materials of construction and methods of fabrication for the miniature aluminum-base irradiation test plates prepared at ORNL and records the pre-irradiation data measured at ORNL.
ORNL-TM-1388 12/8/65	HFIR Cladding-Failure Detector	J. T. DeLorenzo	Describes the operation of the cladding-failure detector and gives a procedure for determining the proper operational settings for the pulse amplifier, pulse-height discriminator, and the high-voltage supply.

ORNL Report No. and Date	Title	Author	Description
ORNL-TM-1393 12/8/65	Description of Facility Radiation and Contamination Alarm Systems Installed in the HFIR Facility, Building 7900	J. A. Russell, Jr. D. J. Knowles	Description of radiation and contamination alarm systems installed in the HFIR.
ORNL-TM-1440 3/9/66	Recommended Hydraulic Measurements in the HFIR Mockup	R. J. Kedl	Recommendations for hydraulic measurements in the HFIR mockup (1/10/61).
ORNL-TM-1471 7/5/66	Determination of the Power vs Reactivity Frequency Response Function of a Power Reactor, with Application to the HFIR	B. R. Lawrence	Space independent numerical technique which avoids evaluation of transfer functions in an algebraic form indicates 100-Mw operation without servo has a frequency response peak at 10 cps.
ORNL-TM-1472 10/24/66	A Mathematical Model for the HFIR Reactivity Calculation	B. R. Lawrence H. P. Danforth J. B. Bullock	Description of a model for calculating expected reactivity changes of the HFIR by means of an on-line digital computer.
ORNL-TM-1488 7/66	Experimental Determination of Safe Handling Procedures for HFIR Fuel Elements Outside the Reactor	S. J. Raffety J. T. Thomas	Experimental determination of safe handling procedures for HFIR fuel elements outside the reactor.
ORNL-TM-1532 Rev. 9/16/66	Operating Safety Limits for the High Flux Isotope Reactor (HFIR) (100 Mw Maximum Power)		Limit established for each operating variable which has direct reactor safety significance.
ORNL-TM-1628 11/66	Fabrication Procedures for Manufacturing HFIR Fuel Elements	J. Binns G. M. Adamson, Jr. R. W. Knight	Procedures for manufacture and inspection of HFIR fuel assemblies.

ORNL Report No. and Date	Title	Author	Description
ORNL-TM-1638 (parts 1 through 38) 1968	Nuclear Instrument Module Maintenance Manuals	J. L. Anderson et al.	Separate manual for each amplifier chassis or component.
ORNL-TM-1677 12/66	In-Place Iodine Filter Tests at the HFIR	J. H. Swanks	Results of in-place iodine filter tests in the SBHE and HOG systems.
ORNL-TM-1687 2/67	HFIR Homogeneity Scanner, Production Model, Operating and Maintenance Manual	J. W. Reynolds R. L. Shipp T. F. Sliski W. H. Longaker K. K. Klindt	Manual describes the operation of the homogeneity scanner and maintenance procedures for the electronic components.
ORNL-TM-1692 2/67	Characterization of U_3O_8 Dispersions in Aluminum	D. O. Hobson C. F. Leitten, Jr.	Characterization of dispersion fuel to help predict irradiation performance and to establish material and process specifications.
ORNL-TM-1712 3/67	The Fabrication of Target Elements for the High Flux Isotope Reactor	J. D. Sease	Covers the fabrication and pertinent preirradiation data of 37 ^{242}Pu-containing target elements to be used for the first HFIR target irradiations, 39 UO_2-Ta-containing simulated target elements for initial startup experiments of the HFIR, and 18 additional ^{242}Pu-containing target elements fabricated earlier for irradiation at SRL.

ORNL Report No. and Date	Title	Author	Description
ORNL-TM-1747 1/18/67	HFIR Transients and Reactivity Accountability	R. D. Cheverton O. W. Burke T. E. Cole	Analog calculations and SPERT comparisons for the final design, for the optimum target-void, indicate that shutdown margins are sufficient and the safety system is adequate for reasonable reactivity insertions.
ORNL-TM-1752 1/17/67	High Flux Isotope Reactor Quarterly Report - July, August, and September of 1966	R. V. McCord B. L. Corbett	Operating report for the third quarter of 1966.
ORNL-TM-1764 2/22/67	Linear-Log Radiation Monitor, ORNL Model Q-2353	J. T. DeLorenzo	Description and parts list of a two-sensor γ ray monitor, effective over 5 decades, for area monitoring.
ORNL-TM-1808 3/20/67	Critical Lattices of HFIR Fuel Elements	E. B. Johnson	Lattices were assembled in order to determine the critical spacing between elements when moderated and reflected by water. Results of these experiments will guide storage of HFIR elements in water, both at the reactor and the recovery plants.
ORNL-TM-1811 3/23/67	High Flux Isotope Reactor Quarterly Report - October, November, and December of 1966	R. V. McCord B. L. Corbett	Operating report for the fourth quarter of 1966.
ORNL-TM-1895 6/9/67	High Flux Isotope Reactor Quarterly Report - January, February, and March of 1967	R. V. McCord B. L. Corbett	Operating report for the first quarter of 1967.

ORNL Report No. and Date	Title	Author	Description
ORNL-TM-1903 12/67	HFIR Fuel Element Steady State Heat Transfer Analysis	N. Hilvety T. G. Chapman	Reports 1960-1964 fuel element steady state heat transfer analysis studies. See ORNL-TM-1904 for later analysis.
ORNL-TM-1904 12/67	HFIR Fuel Element Steady State Heat Transfer Analysis - Revised Version	H. A. McLain	Following tests which showed the fuel element deflects in one direction, a new thermal-hydraulic computer analysis program was written. Power (flow over the entire core instead of just a hot streak) and other changes were made. Analysis showed incipient-boiling power limits were 139 and 153 Mw at 0 and 2500 Mwd, respectively.
ORNL-TM-1961 9/13/67	On-Line Calibration of HFIR Control Rods Using the Rod Oscillation Technique	D. N. Fry	Study made to evaluate the rod oscillation method of measuring differential control rod worth.
ORNL-TM-2017 9/27/67	High Flux Isotope Reactor Quarterly Report - April, May, and June of 1967	R. V. McCord B. L. Corbett	Operating report for the second quarter of 1967.
ORNL-TM-2078 11/13/67	High Flux Isotope Reactor Quarterly Report - July, August, and September of 1967	R. V. McCord B. L. Corbett	Operating report for the third quarter of 1967.
ORNL-2872 3/17/60	Removal of Radioiodine from Air Streams by Activated Charcoal	R. E. Adams W. E. Browning, Jr.	Removal of radioiodine from air streams by activated charcoal.

ORNL Report No. and Date	Title	Author	Description
ORNL-2939 4/29/60	Effect of Heat Flux on the Corrosion of Aluminum by Water. Part I. Experimental Equipment and Preliminary Test Results	J. C. Griess H. C. Savage T. H. Mauney J. L. English	Description of equipment used to conduct experiments on the effect of heat flux on the corrosion of aluminum, and to present the results of the first test.
ORNL-3006 10/4/60	HFIR Preliminary Physics Report	R. D. Cheverton	Preliminary nuclear design of the HFIR.
ORNL-3026 12/20/60	Burnout Heat Fluxes for Low-Pressure Water in Natural Circulation	W. R. Gambill R. D. Bundy	Applicable to after shutdown, natural convection cooling of HFIR elements.
ORNL-3056 2/10/61	Effect of Heat Flux on the Corrosion of Aluminum by Water. Part II. Influence of Water Temperature, Velocity, and pH on Corrosion-Product Formation	J. C. Griess H. C. Savage T. H. Mauney J. L. English J. G. Rainwater	Describes the results of several experiments conducted with 1100 and 6061 aluminum test specimens, which indicate aluminum appears to have a high probability of being a successful cladding material for the HFIR.
ORNL-3063 6/1/61	The Corrosion of Aluminum Alloys in High-Velocity Water at 170 to 290°C	J. L. English L. Rice J. C. Griess	Corrosion tests to determine the behavior of aluminum alloys in high-velocity water under conditions simulating those expected in HFIR.
ORNL-3079 6/5/61	HFIR Heat-Transfer Studies of Turbulent Water Flow in Thin Rectangular Channels	W. R. Gambill R. D. Bundy	Recommended heat transfer correlations for HFIR design (burnout and heat transfer coefficient) with experimental and theoretical justifications.

ORNL Report No. and Date	Title	Author	Description
ORNL-3230 12/5/61	Effect of Heat Flux on the Corrosion of Aluminum by Water. Part III. Final Report on Tests Relative to the High-Flux Isotope Reactor	J. C. Griess H. C. Savage J. G. Rainwater T. H. Mauney J. L. English	Tests to determine the effect of very high heat fluxes on the corrosion of 1100 and 6061 aluminum alloys by water.
ORNL-3290 6/6/62	Transuranium Quarterly Progress Report for Period Ending February 28, 1962	D. E. Ferguson	Status of HFIR target work.
ORNL-3359 5/28/63	Preliminary Solution Critical Experiments for the HFIR	J. K. Fox L. W. Gilley D. W. Magnuson	Experiments were conducted on assemblies with a central H_2O column surrounded by an annular region of uranyl nitrate solution which in turn was surrounded by a heavy and/or light water reflector region.
ORNL-3375 1/18/63	Transuranium Quarterly Progress Report for Period Ending August 31, 1962	Compiled by D. E. Ferguson	Report on the status of the ORNL transuranium work as of August 31, 1962
ORNL-3408 5/31/63	Transuranium Quarterly Progress Report for Period Ending November 30, 1962	Compiled by D. E. Ferguson	Report on the status of the ORNL transuranium work as of November 30, 1962.
ORNL-3442 5/17/63	Tests of High-Efficiency Filters and Filter Installations at ORNL	E. C. Parrish R. W. Schneider	Tests to measure the filtration efficiency of high-efficiency filters.

ORNL Report No. and Date	Title	Author	Description
ORNL-3541 2/64	Effect of Heat Flux on the Corrosion of Aluminum by Water. Part IV. Tests Relative to the Advanced Test Reactor and Correlation with Previous Results	J. C. Griess H. C. Savage J. L. English	Results of tests to investigate the corrosion of 6061 and X-8001 aluminum alloys and the resultant formation of an adherent corrosion product on the corroding surface.
ORNL-3557 2/64	Mechanical Properties of X-8001 and 6061 Aluminum Alloys and Aluminum-Base Fuel Dispersion at Elevated Temperatures	W. R. Martin J. R. Weir	Creep and tensile properties measured over range 70-600°F for ATR fuel plates.
ORNL-3572 Vol. 1 5/64	The High Flux Isotope Reactor, A Functional Description	F. T. Binford E. N. Cramer	Reactor description in support of accident analysis.
ORNL-3572 Vol. 2 8/64	The High Flux Isotope Reactor - Selected Construction Drawings	F. T. Binford E. N. Cramer	Reactor description in support of accident analysis.
ORNL-3572 Vol. 1, Rev. 1 March 1965	The High Flux Isotope Reactor, A Functional Description	F. T. Binford E. N. Cramer	New pages and figures issued to a limited distribution to bring the reactor description up to date.
ORNL-3573 4/67	The High Flux Isotope Reactor - Accident Analysis	F. T. Binford T. E. Cole E. N. Cramer	Includes original accident analysis, and responses to AEC questions.
ORNL-3651 Oct. 1964	Transuranium Quarterly Progress Report for Period Ending February 29, 1964	Compiled by W. D. Burch	Status of transuranium program as of February 29, 1964.
ORNL-3685 9/64	A Reactivity Computer for Use With Nuclear Reactor Control Systems	L. C. Oakes	Describes a computer for checking that regulating rod motion is that expected from reactor's power history.

ORNL Report No. and Date	Title	Author	Description
ORNL-3699 10/64	A Miniaturized Fission Chamber and Preamplifier Assembly (Q-2617) for High Flux Reactors	D. P. Roux E. Fairstein S. H. Hanauer G. C. Guerrant J. L. Kaufman	Description of an articulated assembly consisting of a fission chamber, a preamplifier, and flexible cables, which has been developed to meet the space and environmental requirements of high flux reactors.
ORNL-3737 1/65	Continuous Scanning X-Ray Attenuation Technique for Determining Fuel Inhomogeneities in Dispersion Core Fuel Plates	B. E. Foster S. D. Snyder R. W. McClung	Describes system which automatically scans 0.007 square inch areas.
ORNL-3780 4/65	Development of Nondestructive Testing Techniques for the High Flux Isotope Reactor Fuel Element	R. W. McClung	Report on nondestructive testing methods which were studied, developed, and applied to the evaluation of various properties during fabrication development for HFIR.
ORNL-3848 10/65	Effects of Reactor Operation on HFIR Coolant	G. H. Jenks	Calculations of radiolytic gas production and stability of HNO_3 (added to suppress corrosion film).
ORNL-3929 4/66	Parallel Plate Multisection Ionization Chambers for High Performance Reactors	D. P. Roux	Describes design and testing of HFIR ion chambers.
ORNL-4052 4/67	Characterization and Production of U_3O_8 for the High Flux Isotope Reactor	W. J. Werner J. R. Barkman	Characterization and production of U_3O_8 for the HFIR.

ORNL Report No. and Date	Title	Author	Description
ORNL-4056 1/67	Swelling of UAl_3-Al Compacts	J. L. Gregg R. S. Crouse W. J. Werner	Observations on out-of-pile tests indicate UAl_3-Al compacts swell as UAl_3 is formed, and that vacuum degassing at temperatures near 500°C (below the conversion temperature) or silicon addition may stabilize the intermetallic.
ORNL-4108 10/67	Initial Development of HFIR Fuel Assemblies	R. J. Beaver J. W. Tackett J. H. Erwin G. M. Slaughter W. J. Kucera	Report describes initial development of HFIR-type fuel assemblies aimed at demonstrating feasibility and concluded with the fabrication of the 8-kg-^{235}U aluminum-base fuel assembly for the HFIR critical experiment No. 2
ORNL-4149 10/67	Analysis of Neutron Fluctuation Spectra in the ORR and the HFIR	J. C. Robinson	Analysis of neutron fluctuation spectra in the ORR and the HFIR.
ORNL-4199 1/68	Irradiation Behavior of Aluminum-Base Dispersions Containing Europium Oxides	R. J. Beaver A. E. Richt M. M. Martin	Irradiation behavior of aluminum-base dispersions containing europium oxides.

Appendix D

BIBLIOGRAPHY ON FLUX-TRAP REACTORS

Alichanow, A. I., *et al.*, "A Heavy Water Research Reactor," *Proc. Intern. Conf. Peaceful Uses Atomic Energy, Geneva, 1955,* **2**, 331.

Babcock and Wilcox Company and Lockheed Aircraft Corporation, *Proposal to the Atomic Energy Commission for an Engineering Test Reactor, vol I. Technical Description and Program Plan,* June 30, 1955.

Bolger, J. C., S. F. D'Urso, and R. O. Maak, *Feasibility Study of Superflux Reactor,* EPS-X-293 (December 1956).

Carter, W. L. (group leader), *Design Study of an Advanced Test Reactor* (ORSORT Study), ORNL-CF-57-8-5 (August 1957).

Cheverton, R. D. (group leader), *High Flux Research Reactor* (ORSORT Study), ORNL-CF-56-8-206 (August 1956).

Cheverton, R. D., *Further Studies of a D_2O Moderated High Flux Research Reactor* (to be published).

Choudhury, J., *Critical Calculations for the Core of a Very High Flux Research Reactor,* Geneva Conference Paper 1503, August 1958.

Cole, T. E., and A. M. Weinberg, "Technology of Research Reactors," *Ann. Rev. Nucl. Sci.* **12**, 221 (1962).

Ergen, W. K., *Flux Distribution in a Reactor Consisting of a Spherical Shell of Fuel in an Infinite Moderator,* ORNL-CF-57-12-100 (Dec. 24, 1957).

Ergen, W. K., *Fluxes Obtainable in a Flux-Trap Reactor,* ORNL-CF-58-1-4 (Jan. 15, 1958).

Ergen, W. K., *High Thermal Neutron Flux from Fission. Oversimplified Cases,* ORNL-CF-58-2-127 (Feb. 26, 1958).

Ergen, W. K., *Homogeneous High-Flux Reactor,* ORNL-CF-58-3-68 (Mar. 31, 1958).

Feinberg, S. M., *et al., Uranium–Water Intermediate Reactor Used for Obtaining High Intensity Neutron Fluxes,* Geneva Conference Paper 2142, 1958.

Fowler, T. B., *Maximum Thermal Flux per Mw in Three-Region Homogeneous Reactors,* ORNL-CF-58-3-76 (Mar. 18, 1958).

Francis, T. L., *et al.*, "A Flux-Trap Reactor for Advanced Engineering Tests," *Trans. Am. Nucl. Soc.* **1**(1) (June 1958) (abstract).

Kasten, P. R., M. I. Lundin, and C. L. Segaser, *Aqueous Homogeneous Research Reactor,* ORNL-2256 (April 1957).

Kelber, C. N., *The Theoretical Physics of the Argonaut Reactor,* ANL-5710.

Leyse, C. F., *et al., High Flux Reactors for University Research,* INTERNUC-22 (March 1958).

Leyse, C. F., *et al., An Advanced Engineering Test Reactor,* INTERNUC-23 (see also AECU-3427 and Addendum for preliminary report).

Leyse, C. F., *et al.*, "A High Flux Pool-Type Research Reactor," *Trans. Am. Nucl. Soc.* **1**(1) (June 1958) (abstract).

Link, L. E., *The Mighty Mouse Research Reactor. Preliminary Design Study,* ANL-5688.

Link, L. E., *et al., Design of a High Flux Research Reactor – Mighty Mouse,* Geneva Conference Paper 423, June 1958.

Mallon, R. G., *Conceptual Design of An Advanced Engineering Test Reactor,* ASTRA, 200 E 1.1 (Mar. 1, 1957).

Nestor, C. W., Jr., *Flux Peaking in Homogeneous Reactors,* ORNL-CF-57-8-84 (August 1957).

Osredkar, M., and R. Stephenson, *Thermal Flux in the Central Moderator Zone of Thermal and Fast Reactors,* Geneva Conference Paper 2411, August 1958.

Phillips Petroleum Company, *Cobalt-60 Production Reactor,* IDO-16421.

Ribaric, M., *The Design of Spherical Shell Reactors,* Geneva Conference Paper 475, August 1958.

Stewart, H., *et al.*, "Low-Power Thermal Test Reactor for Nuclear Physics Research," *Nucleonics* **11**(5), 38.

U.S. Atomic Energy Commission, San Francisco Operations Office, *Design Requirements for the Livermore Pool Test Reactor,* May 1955.

INTERNAL DISTRIBUTION

1. Biology Library
2–4. Central Research Library
5–6. ORNL – Y-12 Technical Library
Document Reference Section
7. Laboratory Shift Supervisor
8–41. Laboratory Records Department
42. Laboratory Records, ORNL R.C.
43. G. M. Adamson, Jr.
44. R. G. Affel
45. J. L. Anderson
46. D. S. Asquith
47. S. E. Beall
48. M. Bender
49. F. T. Binford
50–55. A. L. Boch
56. F. R. Bruce
57. C. D. Cagle
58. R. D. Cheverton
59. W. G. Cobb
60–61. T. E. Cole
62. B. L. Corbett
63–87. J. A. Cox
88. E. N. Cramer
89. S. J. Cromer (K-25)
90. F. L. Culler
91. G. L. Dixon
92. B. C. Duggins
93. E. P. Epler
94. W. K. Ergen
95. R. L. Ferguson
96. J. H. Gillette
97. L. A. Haack
98. J. W. Hill, Jr.
99. R. M. Hill, Jr.
100. W. H. Jordan
101. J. A. Lane
102. C. E. Larson
103–127. R. V. McCord
128. H. G. MacPherson
129. H. A. McLain
130. J. R. McWherter
131. L. C. Oakes
132. R. B. Parker
133. A. M. Perry
134. K. H. Poteet
135. M. E. Ramsey
136. A. F. Rupp
137. H. W. Savage
138. T. M. Sims
139. M. J. Skinner
140. C. C. Webster
141. A. M. Weinberg
142. G. Young

EXTERNAL DISTRIBUTION

143. E. E. Hall, DRDT Report Staff, AEC, Washington
144. R. Devine, Bethesda Technical Library, AEC, Washington
145. D. J. Skovholt, Division of Reactor Licensing, AEC, Washington
146. T. R. Jones, Division of Research, AEC, Washington
147. E. E. Kintner, DRDT, AEC, Washington
148. Milton Shaw, Division of Reactor Development and Technology, AEC, Washington
149. M. Biles, Division of Operational Safety, AEC, Washington
150. C. Zabel, ACRS, AEC, Washington
151. S. R. Sapirie, AEC, Oak Ridge
152. H. M. Roth, AEC, Oak Ridge
153. L. H. Jackson, AEC, Oak Ridge
154–155. J. A. Lenhard, AEC, Oak Ridge

156. R. L. Ferguson, AEC, Chicago Operations Office
157. Eric Blomberg, AB Atomenergi, Sweden
158. K. J. Bobin, Mol, Belgium
159. J. R. Colley, South Africa Atomic Energy Board
160. L. F. Franzen, Institut fur Reactorsicherheit der Technische, Koln
161. Masamori Kanbara, Japan Atomic Energy Research Institute
162. F. B. Buoni, Wright-Patterson Air Force Base
163. J. J. M. Snepvangers, Reactor Centrum Nederland
164. Karl E. Wirtz, Karlsruhe, Germany
165. H. Mayer-Leibnitz, Institut Max Von Laue-Paul Langevin, Saclay, France
166–170. J. Chatoux, Institut Max Von Laue-Paul Langevin, Saclay, France
171. N. Hilvety, Union Carbide Corporation, Parma Research Center, Cleveland
172. C. E. Winters, Union Carbide Corporation, Parma Research Center, Cleveland
173. J. A. Swartout, Union Carbide Corporation, New York
174–178. M. Levenson, Argonne National Laboratory, Argonne, Illinois
179. Laboratory and University Division, AEC, ORO
180–436. Given distribution as shown in TID-4500 under Reactor Technology category (25 copies – CFSTI)