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PRELIMINARY HAZARDS SUMMARY REPORT ON THE BROOKHAVEN HIGH FLUX BEAM RESEARCH REACTOR

J.M. HENDRIE AND H.J.C. KOUTS

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BROOKHAVEN NATIONAL LABORATORY

Upton, New York

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BROOKHAVEN HIGH FLUX BEAM RESEARCH REACTOR

by

J. M. Hendrie and H. J. C. Kouts

May, 1961

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1. INTRODUCTION

For several years, greater neutron fluxes than are available from the Graphite Reactor have been needed by Brookhaven scientists to extend existing experimental techniques and develop new experiments, particularly with beams. Physics calculations were, therefore, started in 1956 on a new reactor to provide intense and high quality external neutron beams of various energies from thermal to 10 kev. These studies culminated in early 1958 in the choice of a reactor system which is cooled, moderated, and reflected by heavy water, and in which the core is formed of standard, plate-type fuel elements in a close-packed array (1).

Following the appropriation of design funds for the new reactor, a project group of Brookhaven scientists and engineers was established to work out the preliminary engineering design (2, 3, 4, 5). The Lummus Company was selected in the spring of 1960 to provide architect-engineering services, with assistance from Combustion Engineering on nuclear engineering matters.

A Site Report was prepared and submitted to the AEC in August, 1960 to permit a hazards evaluation of the reactor location (6). The site was reviewed by the AEC and by the Advisory Committee on Reactor Safeguards, and was approved subject to an improved containment of possible radioactive effluents from a reactor accident. These changes have been made in the design of the facility and are reported in this document.

The Lummus Company submitted the HFBR Title I Design Report and Cost Estimate to the AEC in September, 1960 (7). The Title II (detailed design) work has now progressed far enough to permit the preparation of this Preliminary Hazards Summary Report. Actual construction work on the HFBR is scheduled to begin in the summer of 1961.

2. SITE DESCRIPTION

2.1 Reactor Location

The site of the HFBR is shown in Figures 1, 2, and 3, which are maps of the Long Island area around BNL, the BNL South Site, and the populated area of the South Site, respectively. The HFBR is located just east of the Graphite Reactor stack and cooling towers. It uses both of these existing facilities.

The site boundary is closest on the south, where the distance from the HFBR is 3700 feet. On the west, the site boundary distance is 5700 feet; on the north 7200 feet; and on the east 7700 feet. The hospital, which has patients and visitors who are not members of the BNL staff, is 3000 feet from the HFBR. The Laboratory apartment area is 6800 feet from the HFBR.

2.2 Population Distribution

2.2.1 Brookhaven National Laboratory Site Population. The summer (peak) complement of personnel at the Laboratory during working hours is 3,260, with a breakdown as follows:

Regular Laboratory employees	2,380
Temporary employees	50
Guest research scientists	220
Residents of apartment area	225
(families of guest research scientists)	
*Hospitalized patients	30
*Construction contractor employees	215
*Daily visitors	<u>140</u>
Total	3,260

*Daily average.

2.2.2 Population Distribution Off-Site. Brookhaven National Laboratory is located within Brookhaven Town, County of Suffolk, State of New York, as shown on Figure 1. The Brookhaven Town is composed of 413 square miles. It has a population density of 265 persons/sq. mile. The High Flux Beam Reactor is located 13-1/2 miles from the farthest western boundary and 7-3/4 miles from the farthest eastern boundary of Brookhaven Town.

Listed in Table I are the villages and hamlets within a ten-mile radius of the High Flux Beam Reactor and their populations. The population density within this ten-mile radius is estimated at 319 persons/sq. mile.

The major population centers outside the ten-mile radius around the High Flux Beam Reactor are Riverhead to the east, population 5,830, and Port Jefferson to the northwest, population 3,377.

The major portion of the land in Brookhaven Town is used for agriculture, with the remainder being residential areas. The only major industrial or defense facility is the Navy-owned aircraft assembly plant and airfield of the Grumman Aircraft Engr. Corp. at Calverton, 4 miles to the east of BNL.

Brookhaven Town has been an area of increasing population in recent years: on April 1, 1950, the population was 44,522; during the next seven years it increased to 82,459; and in 1960 it was 109,900.

2.3 Meteorology

The Brookhaven area has several meteorological advantages as a reactor site, notably: the absence of any high probability of wind flow toward densely populated areas, and excellent ventilation of the region. In inversion conditions, for instance, the low level winds show a maximum frequency of only 20% from any directional sector. Similarly, 50% of the 300-ft inversion wind speeds are greater than 14 mph.

The Long Island area has relatively few thunderstorms, tornadoes, hailstorms and ice storms, which could increase accident probabilities by disrupting communications and power. The small frequency of such events is attributable to the moderating effect of the nearby ocean, which is, however, responsible for the one detrimental meteorological feature, the hurricane.

While the Laboratory does not have a high hurricane frequency (3 with winds significantly greater than 100 mph since 1900), these storms must be considered in design problems. Fortunately, hurricanes are susceptible enough to existing forecasting techniques for advance warning of several hours to be virtually assured. The HFBR building is designed to withstand a wind speed of 120 mph with an ample safety factor, as are the other local existing reactor structures.

Table I

Populations of Hamlets and Villages within 10 Miles of BNL
as of
April, 1960 (Note 1)

<u>Name</u>	<u>Population</u>	<u>Distance from HFBR</u>	<u>Direction</u>
Ridge	800	1-3/4 Mi.	NNW
Manorville	1000	3-1/4 "	ESE
Middle Island	1400	3-1/2 "	WNW
Yaphank	600 (1000) (Note 2)	3-1/2 "	SW
Moriches	400	4-3/4 "	SSE
Shirley	2000 (8000)	4-3/4 "	SSW
Mastic	1400 (2100)	4-7/8 "	SSE
S. Haven	500	5 "	SSW
Wading River	900 (2000)	5-5/8 "	NNE
Rocky Point	2261	5-3/4 "	NNW
Center Moriches	2521 (3000)	6 "	SE
Coram	400	6-1/4 "	WNW
Shoreham	164	6-1/4 "	N
Calverton	600	6-3/4 "	ENE
Brookhaven	1000	6-7/8 "	SSE
*Wildwood		7 "	NE
East Moriches	1210	7-1/4 "	SE
Mastic Beach	3035 (4000)	7-1/4 "	SSW
Sound Beach	1625 (3000)	7-1/4 "	NNW
Medford Sta.	1800	7-1/4 "	WSW
**Bellport Sta.		7-1/4 "	SW
Eastport	1200	7-3/4 "	SE
Selden	1604 (3000)	8 "	WNW
Hagerman	2000	8 "	SW
Bellport	2461	8-1/4 "	SSW
Miller Place	900 (2000)	8-1/2 "	NNW
Baiting Hollow	500	8-1/2 "	ENE
Speonk	600	8-3/4 "	SE
Mt. Sinai	500 (1000)	9-1/4 "	NW
Farmingville	2134	9-1/4 "	WSW
East Patchogue	5500	9-1/4 "	SW

Table I - continued

<u>Name</u>	<u>Population</u>	<u>Distance from HFBR</u>	<u>Direction</u>
East Patchogue	5500	9-1/4 Mi.	SW
Terryville	800	9-1/4 "	WNW
Holtsville	1500	9-1/4 "	WSW
Remsenberg	600 (900)	9-1/2 "	SE
Patchogue	8838	9-3/4 "	WSW

Note 1. Sources of population data are as follows: Bureau of the Census, Final Population Counts, 1960 Census, PC(AL)-34 Rev., March 16, 1961.

Commercial Atlas & Marketing Guide, 1960 Ed., Rand McNally & Co.

Long Island Lighting Company, Statistical Department, Mineola, N. Y.

Note 2. Figure in parentheses is total population in summer.

* Population included in Wading River figure.

**Population included in Hagerman figure.

The specific location selected for the HFBR, a point southeast of the Graphite Research Reactor, has a number of advantages. Of particular importance is the ability to pipe the ventilation air from the HFBR to the base of the Graphite Research Reactor cooling air stack. This feature assures automatic dilution of the ventilating air by thorough mixing with the large volume of Graphite Reactor cooling air, and it also provides a very high release point for the combined volume. Table II shows the computed effective stack heights achieved by the reactor cooling air under various wind speeds. These are far higher than could possibly be expected from a 350-ft stack serving the HFBR alone. Introducing the ventilating air into the Graphite Reactor stack assures a minimum dilution of 3×10^2 within the stack itself, and an additional factor of approximately 5×10^3 by the time a puff has reached ground level, even under the most adverse conditions.

Table II

Effective Stack Height of the
Graphite Research Reactor Plume

<u>Wind Speed (mph)</u>	<u>Effective Stack Height (ft)*</u> <u>Unstable Conditions</u>
4.5	2570
9.0	1081
15.5	533
22.5	429
29.0	394

*Computed from Bosanquet, Carey & Halton equation, $G = .001$,
3 fan operation.

Table III

Percentage Distribution of 37 ft Wind Speeds

<u>Speed Group (mph)</u>	<u>All Hours</u>	<u>Precipitation Hours</u>
0 - 6	54 %	39 %
7 - 20	45	58
≥ 21	1	3

It is also important to recognize that even if the Graphite Research Reactor were not operating during an accidental release from the HFBR, a natural draft exists, drawing air from the HFBR up the stack rather than allowing it to be released at ground level.

The location of the HFBR takes good advantage of the site area and the natural wind distribution, minimizing the chance that any accidentally released gases or particulates will be carried over the central, populated portion of the site. At the same time good separation from the site boundary is maintained.

Figure 4a, which shows the annual wind roses for both stack level and ground level, indicates that the wind blows toward the site population center during only 25% of all hours. During the remaining 75%, the air moves toward lightly populated areas. The location is also favorable with respect to the off-site population in the directions of the prevailing winds, since the site boundary is about 2 kilometers distant in these directions. This advantage is not derived for winds blowing to the south, where the site boundary is only 1 kilometer distant, but on the other hand this direction includes the region of minimum population off site.

It is important to ascertain whether the wind distribution associated with the worst diffusion conditions is more or less favorable than the general wind rose. Figure 4b answers this question satisfactorily. If a release should occur at ground level during a temperature inversion, there is less probability that the effluent would proceed over the generally populated area of the site than during the better diffusion conditions found during the daytime.

Inspection of Figure 4b does show, however, that a large percentage of inversion hours are recorded as "calms" at the 37 ft level, and the actual direction of air flow during such conditions is in question, since there is usually appreciable air movement, whether recorded or not. Figure 4c shows the general terrain pattern of the area in and about Brookhaven. Smoke diffusion tests conducted at various levels on the Meteorological tower suggest that a northerly drainage wind in near calm conditions is to be expected. We thus feel that the wind rose, Figure 4b, representative of the poorest diffusion conditions, is even more favorable for this location than the statistics show.

In terms of particulate deposition as opposed to cloud dosage, previous experience at this site and the studies of the problem suggest that the least favorable condition would be poor diffusion accompanied by light precipitation (8). Accordingly, the wind rose representation of the hours with precipitation, Figure 4d, is presented. Clearly, this wind rose is less favorable as far as the general site population is concerned, since the northeast, east and southeast winds are represented much more prominently during precipitation hours than is true otherwise. However, this disadvantage is compensated in part by the fact that precipitation situations are frequently associated with higher than average winds, and therefore somewhat better diffusion than normal. This is shown by the comparison of the precipitation wind speed distribution shown in Table III. In the latter, over 50% of the low level winds are found in the 0 to 6 mph grouping, while during precipitation hours, some 58% of the wind speeds range from 7 to 20 mph.

Summarizing the site meteorology, it is felt that the general BNL area is favorable for a reactor site, and that the specific location selected for the HFBR maximizes its safety to both on-site and off-site personnel in every respect except particulate contamination associated with precipitation.

2.4 Hydrology and Geology

The reactor will be located on a small hill of mixed sand, gravel, and clay typical of the Pleistocene glacial deposits which cover the Brookhaven National Laboratory site, and nearly all of central Long Island. The average thickness of the sedimentary outwash and morainal deposit is 200 ft at the reactor site. Various borings have determined the stratigraphy from the surface to bed rock as is shown in Figure 5 (9). Examination of Figure 5 will show that there are variations in the thickness of the several stratigraphic units, as is to be expected in view of the glacial origin of these recent sediments.

In general, there are three definable aquifers between the surface and the impermeable bed rock, which has a depth at the site of about 1600 ft. The upper Pleistocene deposits

constitute the most productive aquifer, the Magothy Formation being somewhat less productive and the Lloyd Sands relatively unproductive in the central Suffolk area. Data on these aquifers are summarized in Table IV, and the flow of ground water through them is shown in Figure 6 (10, 11).

The source of the ground waters underlying Long Island is percolation from rain and snow. It has been estimated that at the BNL site one-third of the percolation supplies the Magothy and Lloyd Formations, while two-thirds enters the Pleistocene stratum. A very small fraction of the ground water in the Pleistocene layer is intercepted by the channel of the upper Peconic River and joins that stream. A still smaller fraction eventually emerges in the Forge River, but much the greatest part of the Pleistocene ground water continues on at the general rate of one-half to one foot per day to form submarine springs in the bays and sea to the southeast of the site. At points near the edges of the Island the hydraulic gradient of the ground water has an upward component and the Lloyd and Magothy Sands will therefore resupply water to the Pleistocene formation.

All irrigation, domestic, process and cooling water at the BNL site is drawn from the Pleistocene and Magothy aquifers, with the Pleistocene supplying much the greater quantity of water. The Pleistocene layer is also the most important water source for water users in the surrounding central Suffolk area.

In the event of a spill of radioactive wastes at or near the HFBR site the possible contamination of surface and ground waters must be considered. The surface of the ground in and near the reactor site is so permeable that flow over the surface for any distance is impossible (12). Following a heavy rain on soaked or frozen ground, small streams flow for short distances, but it would be impossible for contamination to be carried from the Laboratory into the Peconic, Carmans, or Forge Rivers in this way.

The major points in determining the subsequent contamination of the ground water are the direction and rate of movement of the ground water, the dilution of the active solution with the ground water, and the absorption of the

activity by soil materials. The transit time of the Pleistocene layer ground water to the nearest sensitive points outside the Laboratory tract must be measured at least in years and probably in decades, so that only a comparatively few of the longest-lived isotopes are a potential hazard. Because the lamellar flow of ground water precludes mixing by turbulence, dilution should not be of great importance in reducing ground water radioactivity in the event of a spill. It might be expected that much of the ground water stream would remain free of radioactivity but that specific streams of contaminated solution would occur in the layer.

As regards absorption, the environment at BNL is particularly favorable. Even a few hundred feet of a poorly absorbent aquifer like a glacial sand has more than adequate potential absorptive or ion exchange capacity to handle even a very large spill.

Because of the slow and uniform rate of movement of the ground water beneath the Laboratory site, it should be possible in the event of a spill of radioactive wastes to drive scavenging wells to map the limits of the contamination and to pump the active liquid out of the water table. It has been the general conclusion of an extensive study of the geology and hydrology of the Laboratory site that BNL is located in a geologic and hydrologic environment that is unusually well suited to reducing the dangers of serious contamination of the ground water in the event of an accident releasing large quantities of radioactive solution (13).

2.5 Seismology

The probability of occurrence of an earthquake in the BNL area of sufficient intensity to cause damage to building and reactor structures was thoroughly investigated during the construction of the Brookhaven Graphite Reactor. The investigation showed that, in fact, a severe earthquake shock in the Brookhaven area is a most unlikely event. The following paragraph is quoted from the "Supplement to Report on the Brookhaven Nuclear Reactor Prepared for Reactor Safeguard Committee of the Atomic Energy Commission, August 30, 1948:"

Table IV

Average Characteristics of Aquifers and Clay Formations
at the Brookhaven National Laboratory

	Pleistocene Deposits Below Water Table	Gardiner's Clay	Magothy Formation	Raritan Formation Clay	Raritan Formation Lloyd Sand
Thickness in ft	145	10	800	150-200	314
Coefficient of lateral permea- bility gpd/ft ²	1,300	--	100-400	--	74
Transmissibility of whole formation gpd/ft	190,000	--	40,000	--	23,000
Coefficient of vertical permea- bility gpd/ft ²	130-350	0.3	--	--	--
Direction of flow	SE	SE	SE	SE	SE
Undisturbed hori- zontal velocity of flow, ft/day	0.5-1.0	--	.025-0.1	--	0.005
Hydraulic gradient	.0010		.0006		.0002
Porosity, %	35		33		33

Table V

Earthquakes in the Central Long Island Area*

<u>Year</u>	<u>Date</u>	<u>Epicenter</u>		<u>Intensity at Yaphank**</u>
		<u>Lat.</u>	<u>Long.</u>	
1925	Feb. 25	47.6°N	70.1°W	I-III
1929	Nov. 18	44.5°N	55.0°W	I-III
1935	Nov. 1	46.8°N	79.1°W	I-III
1937	July 18	40.7°N	73.7°W	I-III
1944	Sept. 5	45.0°N	74.8°W	I-III
1950	Mar. 29	41.0°N	73.0°W	I-III
1951	Jan. 25	uncertain		not felt

*As reported by the U.S. Coast & Geodetic Survey, Dept. of Commerce.

**Modified Mercalli Intensity Scale of 1931.

"It is, therefore, the consensus of seismologists that no significant quakes are to be expected in the foreseeable future and that, if such quakes are to be expected they will be preceded by preliminary shocks easily detected by seismographs. It is the opinion of the seismologists that the probability of a severe shock would be less than 1 in 100,000 years."

The earthquake history of the site has been reviewed and the principle earthquakes of interest are tabulated in Table V.

The accelerations to be expected from earthquakes of various intensities on the Mercalli scale are shown in Table VI.

Table VI

Earthquake Intensities and Accelerations*

<u>Intensity of Modified</u> <u>Mercalli Scale of 1931</u>	<u>Ave. Maximum Acceleration</u> <u>on One Component</u>	<u>Range of</u> <u>Accelerations</u>
II	2.3 cm/sec ²	1-5 cm/sec ²
III	3.1	1-8
IV	9.3	2-46
V	13.3	2-75
VI	40	5-175
VII	67	18-140
VIII	172	51-350
IX	250	250

*Data from U.S. Coast & Geodetic Survey accelerograph records from 1930 to 1941.

Tables V and VI show that no earthquake has yet produced an intensity in the BNL area in excess of intensity III (1-8 cm/sec²). However, Long Island lies in a Zone 1, or "minor damage" seismic probability area, and for this reason it has been assumed that an earthquake of intensity VII could occur at BNL. The reactor building and all associated structures are, therefore, designed to withstand horizontal accelerations of 0.1 g (98 cm/sec²), which is in the range of an intensity VIII earthquake.

There are no known active earthquake-producing faults in the Long Island area.

3. REACTOR DESIGN

3.1 Design Basis

The purpose of the HFBR is to provide a source of external neutron beams of substantially greater intensity and quality than those now available for fundamental neutron research. Both thermal and epithermal (1-100 ev) beams are of interest. A secondary purpose is to provide some irradiation volumes for high intensity epithermal irradiations and for thermal irradiations.

A theoretical analysis of various reactor systems has shown that the research needs can be met by separating the peak thermal and epithermal fluxes in space, in the reflector and core of the reactor, respectively. This separation is achieved by undermoderating a small, compact core so that most neutrons leak into the reflector before they reach thermal energies. Thermalization then occurs in the reflector, leading to a strong thermal flux peak at some distance outside the core. Thermal beams are extracted from the reflector away from the core, and, therefore, with a much reduced fast neutron background component. Epithermal beams are taken from the core edge, where the epithermal leakage is large.

To save the time and expense of a major fuel element development program, the existing fuel technology has been used. For a small core, fully-enriched uranium fuel in either an aluminum or zirconium structure is indicated. To produce fluxes in the range of interest, the core must run at substantial power densities, and must, therefore, have an efficient cooling system. The coolant choices are water, either H₂O or D₂O, and the liquid metals. The advanced state of water reactor technology over the sodium reactor technology leads us to eliminate the liquid metals. In choosing between H₂O and D₂O as coolant, we chose D₂O to avoid the greater moderating power of H₂O in the core, and the resulting lower leakage of epithermal neutrons from an H₂O system.

With water (i.e., D₂O) chosen as the coolant fluid, and the requirement that the core operate at the highest possible power density, we are led to the ETR fuel element. The ETR currently operates at the highest peak power density of any water reactor. The fuel element which is chosen for the

HFBR, therefore, is an assembly of flat plates, with fully enriched uranium-aluminum fuel sections clad with aluminum. The plate spacings, thicknesses, and lateral dimensions are within the range of ETR practice so that the core hydraulic and heat transfer characteristics are easily extrapolated from ETR experience.

As far as the core is concerned, no further consideration of other moderators, such as beryllium or graphite, is necessary. The moderation provided by D_2O in practical sized fuel cooling channels is all that is required.

The reflector material should permit as large a working volume as possible at high flux in order that there be room for a number of beam tubes and irradiation thimbles in the reflector. Graphite, beryllium, and D_2O have a low neutron absorption and provide a large volume; H_2O does not. Of the three possible reflector materials, the best compromise between high epithermal fluxes, high thermal flux, and ease of handling is D_2O . The over-all choice of D_2O as coolant, moderator, and reflector has many engineering advantages over a system in which several kinds of fluids have to be handled.

3.2 Table of Reactor Parameters

The more important reactor design parameters are collected for convenience in Table VII, on the following page.

3.3 Reactor Physics Design

3.3.1 General. The choices of enriched uranium in aluminum as the fuel, and heavy water as the moderator, reflector, and coolant, are conventional. The arrangement of these materials in the HFBR is, however, unusual, with unusual effects on the neutron behavior.

These effects are most succinctly indicated by the statement that the system is externally moderated. The neutrons liberated during fission in the core spend most of their existence, on the average, in the reflector region. They escape the core generally at a moderately high energy; they are slowed down in the reflector; and the fraction necessary to maintain the chain reaction then diffuse back to be captured at low

TABLE VII. HFBR DESIGN PARAMETER LIST

<u>Power</u>	40 MW	<u>Core thermal design data</u>	
<u>Neutron flux</u>		Water channel thickness	0.105 in.
Core, total epithermal	$\sim 1.6 \times 10^{15} \text{ cm}^{-2} \text{ sec}^{-1}$	Fuel plate thickness	0.050 in.
Reflector thermal flux, max.	$7 \times 10^{14} \text{ cm}^{-2} \text{ sec}^{-1}$	Fuel plate length, active	21 in.
<u>Materials</u>		Water channel width	2.231 in.
Coolant, moderator, and reflector	D ₂ O	Core alloy width	2.00 in.
Fuel	U-235-Al alloy	Heat transfer surface per element	11.1 ft ²
Core structure, beam tubes	6061 aluminum	Total heat transfer surface, core	311 ft ²
Lower vessel	6061 aluminum	<u>Water flow data</u>	
Upper vessel	stainless steel	Water channel flow area per element	0.0326 ft ²
Primary pipes & process equipment	stainless steel	Total channel flow area, core	0.912 ft ²
<u>Fuel element and core</u>		Water velocity in channels	35' ft/sec
Type	ETR, flat-plate	Water flow per element	520 gpm
Uranium concentration in meat alloy	30 wt. %	Water flow, 28 elements	14,560 gpm
Meat thickness	.020 in. U-Al	Water flow in control rods, approx.	2,000 gpm
Cladding thickness	.015 in. 8001 Al	Water flow in bypass, approx.	1,000 gpm
Fuel region dimensions per element	2.6 x 3.0 x 21 in.	Total water flow, approx.	17,600 gpm
U-235 loading per element	260 gm	Total primary loop pressure drop	60 psi
Number of fuel elements	28	Pressure drop in fuel elements	25 psi
Core volume	86 liters	<u>Process system design</u>	
Total U-235 loading	7.28 Kgm	Maximum operating pressure	250 psig
Water-to-metal volume ratio	1.23	Maximum operating temperature	150°F
D/U-235 atom ratio	169	Vessel design temperature	200°F
Al/U-235 atom ratio	123	Beam tube design temperature	400°F
Cycle time for 20% burn-up	40 days	<u>Core power conditions</u>	
<u>Temperature and void coefficients</u>		Radial peak-to-average power ratio	2.15
Core metal coefficient	$-0.36 \times 10^{-3} \% \text{ k}/^{\circ}\text{C}$	Over-all peak-to-average power ratio	2.82
Core water coefficient	$-9.31 \times 10^{-3} \% \text{ k}/^{\circ}\text{C}$	Bulk hot channel factor	1.36
Reflector water coefficient	$-10.5 \times 10^{-3} \% \text{ k}/^{\circ}\text{C}$	Film hot channel factor	1.87
Total temperature coefficient	$-38.0 \times 10^{-3} \% \text{ k}/^{\circ}\text{C}$	Average power density	0.465 MW/L
Core void coefficient	$-0.33 \times 10^{-3} \% \text{ k}/\text{cm}^3$	Maximum power density	1.31 MW/L
<u>Excess reactivity requirements</u>	<u>Maximum</u> <u>Normal</u>	<u>Core thermal analysis results</u>	
Burn-up, plus Sm and stable f.p.'s	5.0 %	Reactor inlet temperature	120°F
Xenon, steady state	3.4 %	Reactor outlet temperature	133.4°F
Temperature	1.2 %	Δ T bulk	13.4°F
Control and experiments	1.1 %	Hot spot surface temperature	360°F
Total	10.7 %	Operating hot spot pressure	175 psig
		Saturation temperature at hot spot	377°F
<u>Control rods</u>		Blanket gas pressure	196 psig
Number of main rods	8	Average heat flux, BTU/hr-ft ²	$.387 \times 10^6$
Individual main rod, total worth	4.1 % k	Maximum heat flux, BTU/hr-ft ²	1.09×10^6
8 main rods in gang, total worth	31 % k	Burn-out heat flux, BTU/hr-ft ²	4.76×10^6
Number of auxiliary rods	8	<u>Neutron prompt lifetime</u>	672 μsec
Individual auxiliary rod, total worth	1.07 % k	<u>Delayed neutron fraction</u>	0.78 % k
8 auxiliary rods in gang, total worth	8 % k		
Total worth of all rods	39 % k		

energy. Not all neutrons pursue this history; some are absorbed in the core before escape, and some which leak out are captured on returning to the core at energies well above thermal. The external moderation is, however, so dominant a feature of the system that the description of the physics behavior is best done assuming that all neutrons take this course.

The neutron physics of the HFBR, including information necessary to establish the effects of structure, the optimum design of experimental facilities, and the core size necessary to support planned operation, has been established in the course of a lengthy set of calculations and experiments (14). Most of the calculations were done on the IBM-704 at New York University. These covered topics such as criticality, flux distributions, the effects of poisons on reactivity, the effect of H_2O contamination of the D_2O , the effectiveness of delayed neutrons, etc. Additional calculations done on the Brookhaven general purpose analog computer have treated kinetic behavior, particularly accident calculations. The experimental program has so far consisted of more than seven hundred critical experiments. These have included generalized studies on the variation of excess k with core size, the reactivity cost of structural components, the variation of beam intensity and quality with beam tube design, power and flux distributions, rod calibrations, some safety studies, the measurement of prompt neutron lifetime, and a number of other topics.

3.3.2 Flux and Power Distributions. Because neutron thermalization takes place predominantly in the reflector, the thermal flux peaks in this region. It is, in fact, this point which makes external moderation so attractive a feature for a reactor designed to produce neutron beams. The beam tubes are located with their inner ends in the reflector, and here they are most efficient in extracting neutrons while at the same time having the least possible influence on reactor criticality. Figure 7 shows a partial breakdown of the space and energy distribution of the neutron flux. The strong reflector thermal flux peak is evident; it can also be seen that this peaking tendency disappears at higher neutron energy. At the highest energy the peak occurs near the core reflector boundary, where the fission rate is greatest.

The power distribution tends to follow the thermal flux curve. Figures 8 and 9 show measured power distributions in Critical Experiment Core 15. Figure 8 shows the vertical average power distribution over one of the symmetrical quarters of the core cross section. Note the peaking along the external corners of the core. Figure 9 shows the vertical traverses for the right-hand peak of Figure 8. The upper graph is the vertical traverse averaged across the outer fuel plate at the corner, and the lower graph is the vertical traverse through the hot spot. The measured peak-to-average ratio is 3.50 in the cold, clean core, and the ratio in the half-dirty core at the beginning of a fuel reloading cycle is measured to be 2.82 ± 0.14 .

3.3.3 Reactivity Coefficients. Because of the external moderation, the reactivity is more sensitive to reflector density change than it is to core temperature changes or core D₂O density. As a result, the uniform temperature coefficient and the void coefficient are not as closely related as is customary.

The void coefficients and temperature coefficient have been found from theory. They are given in the list of design parameters and are

Uniform Temperature Coefficient: $-38.0 \times 10^{-3} \% \text{ k}/^{\circ}\text{C}$
Core Void Coefficient: $-0.33 \times 10^{-3} \% \text{ k}/\text{cm}^3 \text{ void}$.

3.3.4 Reactivity Requirement. The reactivity requirement considered here is the excess which the normal core must have at startup to permit operation for 20 days, or half the 40 day fuel element life. After a normal 20 day operating period the outside 14 fuel elements are to be removed, since they have been in the reactor for two half-cycles, and have experienced the full 20% burnup. The fuel loading plan is to move the inner 14 elements to the outside positions, after which the inner 14 vacancies are to be loaded with fresh fuel. This loading pattern reduces the peak power generation in the hot, outer elements.

Table VIII

Normal Startup Excess Reactivity Requirements
(For 20 Day Cycle)

<u>Source</u>	<u>% k</u>
Equilibrium xenon	3.4
Burnout and stable fission products	2.8
Control and experiments	1.1
Temperature	<u>1.2</u>
Net	8.5 %

The reactivity requirements for a full cycle of 40 days, which would start with a full core of fresh fuel elements, would be greater than that above by 2.2% in k, giving a table as follows:

Table IX

Excess Reactivity Requirements for a 40 Day Cycle

<u>Source</u>	<u>% k</u>
Equilibrium xenon	3.4
Burnout and stable fission products	5.0
Control and experiments	1.1
Temperature	<u>1.2</u>
Net	10.7 %

It will be noted that no requirement is stated for transient xenon or xenon override. The amount of reactivity available for return to operation during a xenon transient will vary depending on when an accidental shutdown occurs within the operating cycle. Just after a half-core reloading, as much as 3.9% k will be available to deal with xenon, this being the sum of the half-cycle burnout allowance and the amount provided for control and experiments. Toward the end of a core half-cycle, the excess k available will be reduced

to the allowance for control and experiments, and will perhaps even be below this level if the central irradiation tubes are in use at the time. If these amounts of k are insufficient to permit catching the xenon transient, the reactor will have to remain subcritical for the 48 hour period of Xe^{135} buildup and decay.

3.3.5 Control Rod Worths and Characteristics. The feature of external moderation leads to very different control and safety rod designs from those customarily found. The most effective position for poison insertion in the HFBR is actually just outside the core, where an absorber can interfere with the return of well thermalized neutrons to the fuel bearing region. As a result, the safety and shutdown rods are all located just outside the core. The rod systems are described in Section 3.5.3. As pointed out there, the HFBR has two sets of eight control rods, making a total of sixteen. Although all are top driven, eight are withdrawn upwards and eight are withdrawn downwards. The former set serve also as safety rods, and on a scram signal these are inserted rapidly into the core by a combination of gravity and spring action.

The reactivity effectiveness of the control rod poison blade to be used in the reactor has been checked out in an intensive set of critical experiments. Calibrations have been performed on individual and ganged insertion of the rods. A typical calibration curve is shown as Figure 10a. It has been determined that there is very little rod interaction of a shadowing nature, and the fraction 0.95 has been deduced as the effectiveness of ganged rods compared to the sum of their individual worths.

The worth of the eight main shutdown rods in ganged insertion is estimated from the critical experiment results to be 31% k . The worth of the eight bottom auxiliary rods is 8% k .

Thus the immediate reactivity insertion on scram is 31% k , and this amount is soon augmented to 39% by the auxiliary rods. The minimum shutdown after a scram is then 21% k through safety rod action, and 29% k after auxiliary rods are inserted. These amounts are, of course, well in excess of any necessary hold-down after a scram.

3.3.6 Delayed Neutron Effectiveness. Since the neutron physics of this system is somewhat unusual, it was necessary to establish the delayed neutron effectiveness, on which the transient behavior depends. This study has been carried through both experimentally and theoretically. The theory has indicated that the system has a delayed neutron effectiveness of 1.05. The experiments have shown that this value is reasonable, and that the usual transient equations are applicable. For conservatism, a delayed neutron effectiveness of 1.0 has been used in transient analysis.

3.3.7 Prompt Neutron Lifetime. Accident analysis requires accurate knowledge of the prompt neutron lifetime. The nature of the HFBR makes it difficult to estimate this quantity theoretically, and so an experimental evaluation was performed.

At an early stage in these studies it was thought possible that the unique nature of the interaction between the reflector and the core might make it necessary to modify the transient equations through assignment of different lifetimes to neutrons absorbed at different energies. The experiments have shown that this is not necessary, the fission by thermal neutrons being well predominant in establishing the chain reaction.

The measured prompt neutron lifetime in a critical mock-up with all structures in place is 672 ± 13 microseconds.

3.3.8 Effects of H₂O Contamination. Every effort has been made in design to make sure that the isotopic purity of the D₂O is maintained during operation. For instance, the overpressure of the primary system establishes the point that heat exchanger leaks would cause loss of D₂O from the primary to the secondary system. Such losses would perhaps be expensive if they were major, but they would not be as costly as would be contamination of the total D₂O inventory by gross amounts of H₂O. Furthermore, they would be more tolerable from the standpoint of reactivity effects, as will now be seen.

The effects of the H₂O contamination of the D₂O are readily understood through the same kind of model as has been used in the previous descriptions, that is, the view of the HFBR as completely externally moderated. The effect of H₂O

contamination of the reflector is to improve the moderating ability, but at the same time because of hydrogen capture to reduce the chance of return of thermal neutrons to the core. The net effect is a reduction in reactivity. If H_2O is added to the core, the reactor tends to better internal moderation, becoming at the limit of infinite H_2O/D_2O ratio a normal water-moderated system of the ETR type. The high uranium loading of the core causes the reactivity to increase monotonically as the H_2O/D_2O ratio increases in the core alone.

If the core and reflector are contaminated by H_2O at the same rate, the dominant effect is first that associated with the reflector. A decrease in reactivity occurs first as this effect sets in strongly. The improved core reactivity begins to be felt after a time; at an H_2O/D_2O ratio of about 0.25 the two effects are about equal, and the net effect is that the reactor has the same reactivity as the purely D_2O moderated system. Beyond this point the addition of H_2O increases the reactivity, until at the limit of pure H_2O in the core and reflector the net effect is a gain in k of about 10%.

The variation of reactivity with H_2O content of the core and reflector separately and together are shown in Figure 10b.

3.4 Reactor Thermal Design

3.4.1 General. The basic feasibility of the HFBR core thermal design is assured by the deliberate similarity of the fuel elements and core assembly to the ETR. As noted in Section 3.1, the HFBR fuel elements are an adaptation of the ETR design, being flat plate, mechanically assembled elements. The plate and water channel thicknesses are essentially the same as those of the ETR, the plate width is slightly less, the plate length is reduced from 37 in. to 22 in. and the uranium loading per unit area is increased. The maximum operating power density and heat flux are slightly lower in the HFBR than in the ETR. The coolant flow rate is set at 35 ft/sec, the ETR value, and flow is downward. In all respects, the HFBR fuel elements at full power conditions are somewhat more conservatively rated than the ETR elements, with which there has been extensive successful experience. In particular, the pressure drop through the HFBR element is reduced by a factor of two below the ETR value because of the shorter plate length, and the plate span is 15% less. Pressure buckling of the fuel plates due to flow differences in adjacent channels should, on these accounts, be much less important a problem in the HFBR.

3.4.2 Flow Requirements. The in-core coolant flow rates based on a 35 ft/sec average channel velocity are listed below:

Water flow per element	520 gpm
Water flow (28 elements)	14,560 gpm
Water flow in control rods, approx.	2,000 gpm
Water flow bypass, approx.	1,000 gpm
Total water flow, approx.	17,600 gpm

The coolant enters the reactor vessel at 120°F and leaves at 133.4°F. The average temperature of the coolant leaving the fuel element is 136°F. This 3° difference is due to the cooling effect of the bypass flow as it mixes with the fuel element effluent.

3.4.3 Hot Channel Factors. There are two methods of combining hot channel factors that arise in reactor thermal design. The first is the cumulative one which assumes that all the individual effects occur simultaneously in the channel exposed to the highest heat flux. The other approach involves a statistical procedure for combining the effects. A design based on the latter has some small but finite probability that local boiling will occur. The former or cumulative method of combining hot channel factors forms the basis for the HFBR thermal design. Table X lists the individual factors involved in the bulk and film temperature rise factors (15).

Table X

HFBR Hot Channel Factors

Variable	Bulk Term	Film Term	Additive
	F_b	F_e	ΔT_{tm}
1. Power density variation	1.10	1.10	
2. Reactor power measurement	1.05	1.05	
3. Channel thickness tolerance	1.05	1.08	
4. Velocity variation, channel to channel	1.10	1.08	
5. Velocity variation within a channel	1.00	1.05	
6. Fuel core alloy variation	1.02	1.10	
7. Heat transfer coefficient	1.00	1.20	
3. Pressure and inlet water temperature measurement	-	-	20°F
Total	1.36	1.87	20°F

3.4.4 Hot Spot Temperature Calculation. The metal surface temperature at any point in the hot channel of the core is given by:

$$T_{HC}^S = T_O^B + \bar{p} \left[\frac{F_C F_b F_r(z) (t_w + t_p) w_m L}{V \rho C_p t_w w_c} + \frac{F_h F_\theta F_o(z) (t_w + t_p)}{2 h} \right] + \Delta T_{tm}$$

where

C_p = specific heat of coolant, 1.0 BTU/lb-°F

F_b = hot channel factor for bulk rise, 1.36

F_θ = hot channel factor for film rise, 1.87

F_C = fraction of fission energy appearing as thermal energy in coolant, 0.95

F_h = fraction of fission energy transferred to coolant by convection, 0.88

$$F_r(z) = \int_0^z p(z, \bar{R}) dz / L \bar{p} = 2.15 \text{ when } z = L$$

$$F_o(z) = p(z, R) / \bar{p} = 2.82 \text{ when } z = L$$

h = heat transfer coefficient = 11,000 BTU/hr-ft²-°F at hot spot

L = length of fuel plate, 21 in.

$p(z, \bar{R})$ = fission power density averaged over width of hottest fuel plate at axial position z (based on volume of fuel plates and water channels)

$p(z, R)$ = local fission power density in hottest fuel plate at axial position z

$\bar{p}$ = average fission power density, based on volume of fuel plates and water channels, 0.708 MW/L

R = radial position of hottest fuel plate

t_w = water channel thickness, 0.105 in.

t_p = fuel plate thickness, 0.050 in.

T_O^B = bulk coolant inlet temperature, 120°F

V = average velocity in water channel, 35 ft/sec

w_m = meat width, 2.000 in.

w_c = water channel width, 2.231 in.

ρ = coolant density, 68.3 lbs/ft³

z = axial position measured from top of fuel section
(note that this z convention differs from that of
Section 3.3.2 and Figure 9, where $z = 0$ is taken at
the center of the core).

Since the power density distribution is approximately symmetrical about the horizontal midplane, the hot spot will lie somewhere between the middle and the outlet of the hot channel. The calculated surface temperatures in this region are found to vary about 5°F, with the maximum value (360°F) occurring at the channel outlet, i.e., where $z = L$. The system is pressurized so that the hot spot operating pressure is 175 psig, with a corresponding saturation temperature for D₂O of 377°F.

3.4.5 Steady State Burnout Analysis. A survey of existing methods of burnout prediction (15) reveals that the Mirshak (16) formula is based on data which comes closest to matching the operating conditions of the HFBR. The only variable which falls outside the range of the Mirshak data is coolant pressure (~ 190 psia in HFBR and 25-85 psia in the experiments). The Mirshak correlation is displayed graphically in Figure 11 for the pressure range of 25-85 psia. The stated channel geometry, flow rate and temperature conditions are those of the HFBR.

Bernath (17) and Menegus (18) have developed more general burnout formulae which have been tested against data representing a wide range of pressures, heat fluxes, coolant flow rates and channel geometries. These equations have been plotted along with the Mirshak correlation in Figure 11. A factor of 1.4 is included here to reflect the experimental deviation.

Of the three curves, the Mirshak correlation can be assumed to be the most reliable in the pressure range 25 to 85 psia. Furthermore, the wide range correlation which best approximates the Mirshak line in this interval is probably more suited to estimating burnout at higher pressures. Examination of Figure 11 shows that the Bernath equation has these characteristics and consequently is chosen as the best equation for HFBR conditions.

The Bernath correlation is given as

$$q_{BO} = \frac{h_{BO}}{1.4} (T_w - T_b)_{BO}$$

where

$$h_{BO} = 10,890 \left(\frac{De}{De+Di} \right) + \frac{48}{De^{0.6}} (V)$$

$$T_w = 57 \ln P - 54 \left(\frac{P}{P+15} \right) - V/4$$

The terms are defined as follows:

q_{BO} = heat flux at burnout, PCU/hr-ft²

De = hydraulic diameter, ft

Di = "heated" diameter; heated perimeter divided by π , ft

V = coolant velocity, ft/sec

P = pressure, psia

T_b = bulk temperature at hot spot, °C

1.4 = safety factor.

The burnout heat flux may be calculated from the above equations on several bases. First, it is useful to compare the operating maximum heat flux with the burnout heat flux calculated for the operating temperature and pressure conditions in the bulk water in the cooling channel. For this case the burnout heat flux is 4.76×10^6 BTU/hr-ft² compared to the maximum operating value of 1.09×10^6 BTU/hr-ft². A second approach is to assume a slow increase in reactor power, with the inlet water temperature held constant. In this case the bulk water temperature at the hot spot is raised, and the reduced subcooling at the hot spot reduces the burnout to 3.26×10^6 BTU/hr-ft². It should be noted that the burnout calculations above assume that all of the variable factors in the thermal design will have their "worst" possible values occur simultaneously at the point of maximum power generation.

3.4.6 System Pressures. The pressure losses in the HFBR core have been calculated from standard formulae and the results checked with the measured values from ETR element

hydraulic tests. The total frictional loss across the core is 20.9 psi at the normal flow velocity of 35 ft/sec in the coolant channels. Additional losses in the remainder of the water circuit within the vessel add to 8.5 psi, and give a total pressure loss of 29.4 psi from inlet to outlet of the reactor vessel (19).

The primary piping and heat exchangers have a total frictional loss of 23 psi; this, together with the vessel drop and an allowance of 7.6 psi across the main throttle valves give a total primary system pressure drop of 60 psi at the normal flow rate. System pressures are established on the basis of a cover gas pressure of 196 psig, which leads to ample pressurization at the core hot spots to suppress boiling (20).

3.4.7 Gamma Heating. The gamma heating to be expected in and around the reactor core has been calculated and measured. The results are shown in Figure 12, which is taken from Critical Experiment Report No. 20 (14). The comparison curve marked "BSF" in Figure 12 is the measured gamma heating rate in light water surrounding the Oak Ridge Bulk Shielding Reactor.

3.4.8 Control Rod Heating. Aside from the fuel elements themselves, the most critical heating problem in the reactor is encountered in the control rods. The control blades are sometimes close to the core, in the strong gamma field, and, in addition, capture many neutrons with resultant energy generation. The shape of the curve of neutron current into a control blade as a function of distance from the end of the blade has been measured. The results are given in Figure 13. The poison materials contemplated for the blades are cadmium or dysprosium, both of which yield fairly energetic gamma rays following neutron capture. The control blade heating is due then to both the core gamma flux and to the gamma flux from neutron absorption in the blade. The partial and total heating effects are shown in Figure 14 (21).

The maximum heat flux of 230,000 BTU/hr-ft² would be transferred to the water by the two surfaces of a solid blade, or the four surfaces of a blade with an internal water cooling channel. The cooling of the blade may be accomplished either by natural convection or by forced flow. Natural convection cooling to the bulk water in the reactor vessel would occur with boiling on the blade surface. The metal surface temperature in this case would be reasonable for stainless steel clad rods, being about 420°F (saturation temperature in

the region of the rods is about 370°F). The heat flux would not be high enough to cause concern about burnout. The boiling would, however, cause neutron flux variations, and this flux "noise" is considered undesirable in a reactor for neutron beam experiments. Forced flow cooling through an internal channel has been adapted, therefore, to prevent boiling at the hot ends of the blades. Should the forced flow cooling fail, due to plugging of the internal water channel, the blade will revert to boiling and natural convection cooling with no excessive temperature rise.

3.4.9 Fission Product Afterheat. The total heat production due to fission products for the hot element and for the entire HFBR core is given in Figures 15, 16, and 17 as a function of time after shutdown from 40 MW power, after a normal cycle.

3.4.10 Primary Coolant Coastdown. In normal operation of the reactor plant the primary coolant flow is established well before the reactor is brought to power. Similarly, the full coolant flow is maintained for some time after the reactor is shut down. It is necessary to consider, however, those abnormal conditions in which a failure of electrical power to the plant causes the main pumps to shut down simultaneously with the reactor. The matters to be considered are the coastdown time of the primary coolant to the shutdown cooling flow level, and whether or not the core metal temperatures reach the saturation temperature during the transient.

The HFBR primary loop has two parallel pump and exchanger legs, and the flow coastdown has been calculated for a) inertia of two pumps and motors; b) inertia of one pump and motor; and c) inertia of the primary fluid alone (22). The results are given in Table XI.

The shutdown cooling flow rate is 500 gpm, or 0.028 of full flow, so that the shutdown cooling flow is reached in about 100 seconds for case a), 74 seconds in case b) and 36 seconds in case c).

The fuel plate maximum surface temperature has been calculated for the coastdown period. The maximum temperature occurs at the least flow, i.e., at the shutdown cooling flow level and is about 100°F below the saturation temperature (23).

Table XI

<u>Primary Coolant Coastdown Times</u>			
<u>Fraction of Full Primary Flow</u>			
<u>Time After Power Failure</u>	<u>a) including inertia of two pumps</u>	<u>b) including inertia of one pump</u>	<u>c) neglecting inertia of pumps</u>
1 sec	0.76	0.67	0.51
5	.38	.29	.175
10	.24	.175	.095
20	.14	.095	.049
40	.072	.050	.025
60	.050	.034	
80	.038	.026	
100	.031		

3.4.11 Shutdown Cooling Flow. The shutdown cooling flow rate is 500 gpm. This flow is provided by either of two parallel shutdown pumps and a separate heat exchanger. The shutdown pumps have three sources of electrical power; the normal supply, an emergency power generator set driven by a propane engine, and a storage battery bank. One of the shutdown pumps is in operation at all times, so that on a failure of the main pumps the primary coolant flow coasts down to the 500 gpm shutdown level with no action required of the operators.

At the 500 gpm level, the fuel plate surface temperatures are well below the saturation temperature (23). The fuel temperatures reach a maximum as the primary flow slows down to the 500 gpm level, and decrease slowly thereafter as the fission product afterheat decreases.

The stability of downflow at the 500 gpm level through the core has been examined to see if there is any danger of internal circulation, i.e., of upflow in the hotter channels and downflow in the cooler channels. There is, in fact, no such tendency and the shutdown flow will be stable and well distributed across the core (24).

3.4.12 Natural Circulation Cooling. In the event of failure of all forced cooling flow through the reactor core, the fission product afterheat must be removed by the natural thermal circulation of the heavy water within the lower portion of the vessel. The HFBR vessel and shield arrangements are such as to insure the core being covered with heavy water in all cases of pipe breaks and vessel leaks.

Since a means for bypassing heavy water through the flow shrouds at the top of the core is provided (the emergency bypass orifices, shown in Figure 18), a thermal siphon loop will operate within the lower portion of the reactor vessel. In this circuit, flow is upward through the core. At times soon after reactor shutdown, when the afterheat is large, boiling in the core provides the driving force for the flow. At long times after shutdown, with smaller afterheat, the circulation is sustained by the density difference in the hot, in-core leg and the colder, exterior leg of the circuit, just as in the natural cooling of used fuel elements in the storage canal. The critical period is during that time in which the afterheat is sufficient to cause boiling, and these remarks will henceforward be confined to that condition.

The afterheat is removed from the core by sensible heat and heat of vaporization of the heavy water. At the top of the core steam disengages from the steam-water mixture and rises into the upper part of the reactor vessel. The water flows through the bypass orifices into the reflector region of the vessel, and is returned to the bottom of the core. The thermal siphon flow is opposed by frictional losses in the nucleate and bulk boiling regions in the core, by acceleration losses from vaporization, and by the frictional loss in the bypass orifices. If the driving force is sufficiently large compared to these losses, then the flow rate will be large enough to prevent burnout in the hot channels.

Two burnout flow rates are of interest: that for the average channel in the core, and that for the hottest channel. The total flow in the circuit (and the design of the bypass orifices) are determined by the heating rate in the average channel, and the minimum permissible flow rate corresponds to the burnout flow rate in the average channel. Stable operation of the circuit at a given total flow rate is then assured if, at the same time, the flow rate in the hot channel exceeds the burnout flow rate there.

The thermal siphon flow rates in the average channel and in the hot channel have been calculated for several pressure and inlet temperature conditions (25), and are given in the first three columns of Table XII. The heat fluxes in the average and hot channel were taken as 16,400 and 49,300 BTU/hr-ft², respectively, which were current values when the calculation was made. (Subsequent new information on the power distribution and recalculation of the afterheat distribution has given 8,000 and 19,000 BTU/hr-ft² for the average and hot channel peak heat fluxes (26). The thermal siphon circuit has not been recalculated as yet, but the approach to burnout will be markedly reduced by the new numbers.)

It is clear from Table XII that a reduction in the system pressure to atmospheric at the top of the heavy water column increases the flow rates. The effect is that at lower pressures boiling occurs earlier in the heated channels, and since the driving force is due to the reduced density of the steam-water mixture, more boiling means more driving force. Accordingly, the vessel relief valves are arranged to open on the same safety signal which opens the bypass orifices, and thus reduce the system pressure to atmospheric at the start of the natural circulation process.

Table XII

Thermal Siphon Circuit Conditions

(for 16,400 and 49,300 BTU/hr-ft² average and hot channel heat fluxes)

A. Average Channel

<u>Core Pressure</u>	<u>Inlet Temperature</u>	<u>Flow Rate W</u>	<u>Burnout Flow Rate</u>	<u>W/W_{BO}</u>
			<u>W_{BO}</u>	
17.5 psia	120°F	85 lb/hr	20 lb/hr	4.2
17.5	200	251	20	12.5
224	120	33	20	1.6
224	370	355	20	17.8

B. Hot Channel

224	120	99	70	1.4
17.5	120	(200-250 estimated)	70	(~3)

Data are available on the burnout flow rates in heated vertical channels with natural circulation at atmospheric pressure. A summary of the data is given in Figure 19, which shows that the average and hot channel burnout flow rates for the 16,400 and 49,300 BTU/hr-ft² values are 20 and 70 lbs/hr, respectively. (At the newer heat fluxes of 8,000 and 19,000 BTU/hr-ft², the burnout flow rates are reduced to 9 and 25 lbs/hr.)

For the atmospheric pressure (17.5 psia) cases, the margin to burnout for the thermal siphon circulation is ample, and the core will be adequately cooled. Since some of the steam evolved passes out the open relief valve, the heavy water level will slowly fall in the vessel. A source of makeup heavy water is provided on the Operations Floor, with a gravity feed to the reactor vessel.

The remaining question in regard to natural circulation cooling of the core involves the process of flow reversal from the forced downflow to the thermal siphon upflow. The opening of the bypass orifices during the coastdown period should assist the reversal, as should the venting to atmospheric pressure which will speed up the onset of boiling. We believe that the flow reversal will take place without difficulty, and have started a series of experiments to examine the reversal process in detail.

3.4.13 Heat Removal in Fuel Handling. The spent HFBR fuel elements are removed from the heavy water in the reactor vessel and are transferred in air to the light water storage canal. The elements are adequately cooled in static water either in the vessel or the canal. The transfer through air, on the other hand, requires forced flow cooling if the element is to be in air for longer than about five minutes. The cooling method is to pass some 200 cfm of room air through the element water channels. After the normal cooling period before fuel is removed, the average heat flux is 2750 BTU/hr-ft². The cooling air flow velocity is 88 ft/sec (the pressure drop is about 20 in. of water) and the peak fuel temperature is 366°F (27). The average fuel temperature is about 100° less: 267°F.

3.4.14 Thermal Shield Heat Removal. The thermal shield is a right circular cylinder of steel and lead-cadmium eutectic surrounding the spherical portion of the reactor vessel. It absorbs the thermal neutron leakage, the core gamma leakage and the capture gammas from the neutron leakage and protects the concrete shield from these radiations. The concrete biological shield outside the thermal shield will absorb a maximum energy current of about 3×10^{11} Mev/cm²-sec without danger of cracking from internal thermal stresses.

The centerline gamma flux at the thermal biological shield interface has been calculated to be 2.11×10^8 Mev/cm²-sec-MW, or 8.44×10^9 Mev/cm²-sec at 40 MW reactor power (28). The thermal neutron energy current into the concrete is essentially zero, and the fast neutron energy current will lie between 0.2 and 1×10^{11} Mev/cm²-sec, including the capture energy.

The thermal shield wall is composed of the following sections, from inside to outside:

- A. 3/4 in. steel plate
- B. 1/2 in. cooling water gap
- C. 1/4 in. steel plate flow baffle
- D. 3/4 in. cooling water gap
- E. 1 in. steel plate
- F. 4-1/2 in. lead-cadmium eutectic
- G. 1 in. steel plate.

The heat generation and internal temperature variations in the various solid sections of the thermal shield are given in Table XIII for the design cooling flow conditions.

Table XIII

Centerline Heat Generation and Internal Temperature Variation in the Thermal Shield Wall for 40 MW Power Level

<u>Wall Section</u>	<u>Heat Generation per Unit Face Area</u>	<u>Temperature Drop Across Section</u>
A	11,440 BTU/hr-ft ²	19.7°F
C	2,080	0.9 max.
E	4,680	17.9
F	3,760	
G	0	5.4
Total	21,960 BTU/hr-ft ²	

The distribution of the thermal shield heat load and the total load is given in Table XIV.

Table XIV

Thermal Shield Heat Load Distribution

Cylindrical Side Wall	3,010,000 BTU/hr
Bottom	625,000
Top	<u>300,000</u>
Total	3,935,000 BTU/hr
Per cent of Reactor Power	2.9

3.5 Reactor Core

3.5.1 Fuel Elements. The HFBR fuel element, shown in Figure 20, is a flat plate, ETR-type aluminum-uranium assembly. Each element contains 19 fuel plates, .050 in. thick, 2.231 in. wide, and 22 in. long. Water channels of .105 in. between fuel plates are maintained by mechanically attaching the fuel plates to aluminum side pieces. The complete assembly is 40 in. long including the handling ears at the top ends of the side pieces. A 14 in. space between the lattice plate and the bottom of the fuel plates provides a reflector volume immediately below the fuel bearing section of the element.

Each element contains 260 grams of U-235 distributed uniformly throughout the meat sections of the 19 fuel plates. The alloying material in the meat section is 1100 aluminum. X-8001 aluminum alloy is used as the cladding on the surface of the plates. The meat section is .020 in. thick with .015 in. of cladding metal on either side. The uranium concentration in the meat alloy is 30 wt %, with some silicon added to the alloy to obtain more uniform distribution of the uranium-aluminum compound.

The fuel plate is fabricated from the meat section, a picture frame section surrounding the edges of the meat, and two cladding plates. These components are bonded together

by the hot rolling process with subsequent cold rolling to achieve the final plate thickness. The fuel plates are attached to the side pieces by a mechanical assembly method rather than brazing, which tends to soften the plates. The inspection procedure for individual fuel plates will include a blister test, radiographic inspection for uranium segregation, and an ultrasonic test to locate unbonded areas. For the completed elements a hydraulic test at twice the normal pressure drop will check the strength of the assembly. Specific items of interest in the hydraulic test will be the buckling or bowing of any fuel plates and over-all warpage of the element.

A test program has been conducted on prototype fuel elements to determine the degree of bonding achieved between the meat section and the X-8001 aluminum alloy cladding. The results show that X-8001 aluminum alloy is no more difficult to roll-clad than 1100 aluminum. The radiation stability of elements with 30, 35, and 40 wt % meat alloys is currently being tested in the ETR. The test involves irradiation of prototype elements to 40% uranium burnup with subsequent hot cell examination. The hot cell measurements will include metallographic examinations, burnup analysis, and dimensional checks. These test elements are clad with X-8001 aluminum.

3.5.2 Core Structure. The core mechanical structure includes a lattice plate, a lattice plate mounting ring, and a shroud assembly. The lattice plate and shroud support 28 fuel elements and 2 in-core irradiation tubes. The arrangement in the reactor vessel is shown in the horizontal cross section of Figure 21. An isometric view of the core shroud structure is shown in Figure 22. The shroud assembly directs the flow of coolant water through the fuel elements. The downward thrust of the fuel elements due to the water flow through the core is transmitted to the lattice plate, the lattice plate mounting ring, and to the base of the reactor vessel through a welded aluminum support structure.

The lattice plate, located 25 in. below the core centerline, is fabricated from a stainless steel forging. It is 28-1/2 in. in diameter, 4-1/2 in. thick and has a shoulder machined to fit the inside of the shroud. The shroud overlaps

the shoulder and can slide to adjust for differential thermal expansion. Other openings in the lattice plate include 2 in. diameter holes below each fuel element, through which a portion of the coolant flows into the reflector, and apertures to guide the lower set of control rods. The lattice plate is securely bolted to, and aligned with the mounting ring mentioned above.

Beneath the lattice plate a series of radial, stainless steel plates direct the flow of heavy water into the reflector volume. These steel elements, together with the lattice plate, form a "critical-safe" region in the bottom of the reactor vessel where fuel plate fragments may collect without any criticality hazard.

The shroud assembly consists of a stainless steel cylinder to which is fastened an aluminum lower section that extends through the core region to the lattice plate below. The assembly is shown in Figure 25, and the shroud details can be found in Figure 30.

The upper shroud is 32 in. O.D. x 3/4 in. wall and is flanged at its top for bolting to a support ring in the neck of the reactor vessel. At its lower end, 14 in. above the core, is located a 2 in. thick transition plate upon which the lower shroud is mounted. The upper set of control rods operate through openings in the transition plate. There are additional openings in the transition plate for two core-edge irradiation tubes and four emergency bypass valves. Both upper shroud and transition plate are of type 304 stainless steel.

The lower shroud is fabricated from 6061-T6 aluminum alloy. A number of 1/8 in. thick partitions divide the core area into 30 fuel element boxes and serve as tension members between the 3/16 in. thick perimeter plates. Below the core, part of the coolant escapes to the reflector region through a number of holes in the perimeter wall. The lower shroud is bolted to the transition plate and can be replaced, if necessary, without discharging the stainless upper section.

3.5.3 Control Rods. The control rod configuration adopted for the HFBR is shown in Figure 23. There are a total of 16 rods, divided into 8 main rods operating above

the core and 8 auxiliary rods operating below the core. The rod cross sections are angles, both because this section gives good mechanical strength to the blades and because the angle shape is a good fit to the core outline. The safety requirements are adequately met by action of the main rods alone, so that only the main rods are dropped on a scram signal. In the fully inserted position the main rods extend along the sides of the core to a point 4 in. from the bottom. The auxiliary rods in the fully inserted position extend 2 in. up along the core. To avoid the difficulties inherent in bottom penetrations of the reactor vessel for control rod drives, both the main and auxiliary rods are actuated by drive mechanisms high on the vessel. Both the main rod blades and the followers which connect the auxiliary blades to their drive mechanisms pass through the transition plate above the core. This geometry, together with a cross section through the rod blades is shown in Figure 24.

The reactor is operated with the two sets of rods in symmetrical positions above and below the core centerline. This insures a minimum flux perturbation in the horizontal midplane of the reactor and hence a relatively constant source flux for the beam tubes. The main and auxiliary rod poison sections have identical reactivity characteristics over the pertinent control range. Since the main rods have an individual worth of 4.1% in k each, any two of the eight main rods will shut the reactor down after it has reached normal operating temperature and equilibrium xenon. Any three of the eight main rods will shut down the reactor in any configuration. The total shutdown worth in the eight main rods is 31% in k , and in the eight auxiliary rods 8% in k . The auxiliary rods are driven to the full "in" position at normal speed on a scram signal. They are not needed for safety reasons, but the automatic insertion insures identical rod positions for all startups.

The control rod blades are spaced 1/2 in. outside the lower shroud. The main rod blades are rigidly attached to heavy stainless steel follower tubes and are not otherwise guided below the transition plate. The auxiliary rod blades, which are attached to relatively flexible aluminum follower tubes, are fully restrained and guided by appropriate structural members below the core.

The normal, motor-driven rod velocity is 4 in./min for both insertion and withdrawal of all rods. Manual control of the rods is in one of three modes: motion of any single rod, motion of all eight main rods in gang, and motion of all eight auxiliary rods in gang. The 4 in./min rod velocity leads to a reactivity rate around the critical position of at most 8 cents/sec for an eight rod gang. Automatic control is limited to motion of a single auxiliary rod.

The specifications on the main rod fast drop require that the drop result in -67 cents in reactivity per rod within 180 milliseconds of the occurrence of an unsafe reactor flux or process condition. For the eight main rods this corresponds to -4% in k in 180 milliseconds. The required acceleration to meet this specification is about $3 g$ (29).

The control rod drive mechanisms will be similar to the reference design mechanism shown in Figure 29. There is an individual drive unit for each rod. The main rod blades are attached to heavy stainless steel followers which slide in bearings mounted in stationary support tubes. As a rod is withdrawn a heavy spring is compressed to provide the energy storage for the required fast drop acceleration. The drive is a rack and pinion type with an electromagnetic decoupling clutch outside the reactor vessel.

The auxiliary rod blades are attached to aluminum followers which slide in bearings above the transition plate. The auxiliary rod blades slide within guides below the core. The control rod blades are constructed of stainless steel, with either cadmium or a stainless steel-dysprosium cermet as the poison material.

3.6 Reactor Vessel

3.6.1 General. The core is contained in the reactor vessel, which is shown in Figures 18, 25, 26, 27, and 28. It consists of a pressure vessel, various vessel internals, and a support skirt. The vessel has an over-all height of about 24 ft. It is designed in accordance with the ASME Unfired Pressure Vessel Code for the most severe condition of coincident pressure and temperature expected in normal operation, conservatively selected as 275 psig and 200°F. Applicable Code formulas are supplemented, where required, to account for the thermal stresses arising from radiation absorption heat production.

3.6.2 Lower Pressure Vessel Section. The lower pressure vessel section consists of a spherical shell having a 4 ft O.D. cylindrical neck at the top. The section has an over-all height of about 9 ft, an outside diameter of 7 ft, and a minimum allowable wall thickness of 1-1/16 in. The calculated thermal stress for this thickness has a maximum value of 240 psi. The minimum allowable wall thickness of the cylindrical neck is 1-3/16 in. This section of the vessel is constructed of 6061-T6 aluminum. The normal operating pressure is 200 psig or less, and the heavy water reflector maintains the vessel wall temperature below 200°F at reactor powers up to 100 MW (30).

3.6.3 Beam Tubes and Thimbles. The lower pressure vessel section has welded into it a series of beam tubes and reflector irradiation thimbles as shown in Figures 25, 26, and 28. There are eight beam tubes and three thimbles with a 3-1/2 in. I.D. and one beam tube with a 12 in. I.D. The latter beam tube is a cold neutron facility. Both the beam tubes and the thimbles are designed for 275 psig external pressure and 400°F, except that in the calculations for the reinforcement of shell openings a temperature of 200°F is used. The tubes are constructed of 6061-T6 aluminum.

The tubes are cooled by the primary coolant flow through the reflector. The two thimbles V-10 and V-11 have a minimum allowable wall of 3/8 in.; the hemispherical heads have a minimum allowable wall thickness of 5/32 in. They are maintained at a temperature of 300°F or less, up to reactor powers of 100 MW (31).

The eight 3-1/2 in. I.D. beam tubes and the thimble V-12 have a relatively high radiation volumetric heating rate due to their close approach to the core. To provide adequate surface for the high heat fluxes, they are constructed with triangular profile cylindrical fins machined in their walls. These fins also act as stiffening rings and allow the tube wall thickness to be reduced to 3/16 in. The hemispherical heads have a wall thickness of 1/8 in. and have fins for heat transfer surface. The beam tubes are maintained at a temperature of 300°F or less up to reactor powers of 80 to 100 MW (31).

The 12 in. I.D. beam tube has a minimum allowable wall thickness of 1-5/16 in., and hemispherical head wall of 1/2 in. The tube is maintained at a temperature of 300°F or less, up to reactor powers of 83 MW, and at a temperature of 350°F or less, up to reactor powers of 100 MW.

3.6.4 Lattice Plate Support Structure. The stainless steel (type 304) lattice plate is bolted to an aluminum support ring which, in turn, is welded to two perpendicular aluminum support webs. The webs are welded directly to the vessel wall and transmit the static and dynamic forces from the support ring to a wide area of the vessel wall. Both the ring and the support webs are 6061-T6 aluminum. An array of stainless steel plates bolted to the support webs directs the coolant flow up into the reflector.

3.6.5 Radiation Heating in Structures. The assumed maximum volumetric radiation absorption heating rates of the various structural components are tabulated in Table XV.

3.6.6 Upper Pressure Vessel Section. The upper pressure vessel section is flanged to the lower section and consists of a cylindrical shell with a stepped, flat-plate head arrangement at the top. The section has an over-all height of about 15 ft, an O.D. of 4 ft, and a minimum allowable wall thickness of 13/32 in. This section is constructed of SA167, Type 304 stainless steel. The normal operating pressure and temperature are 200 psig and 150°F, respectively.

The stepped, flat-plate head arrangement serves a number of useful purposes; it allows the in-core and core edge irradiation thimble cooling water connections to enter the thimbles below the service floor, relaxes the required bending radius of the thimbles as they pass from the vessel to

Table XV.

Assumed Radiation Heating Rates in Internal Structures

<u>Structure</u>	<u>Identi- fication No.</u>	<u>Distance from Core Edge, cm</u>	<u>Maximum Volumetric Heating Rate/MW Power, watts/cm³-MW</u>
Beam Tubes	H-1	4.6	0.783
	H-2	6.1	0.689
	H-3	14.6	0.373
	H-4	15.2	0.356
	H-5	16.0	0.340
	H-6	14.5	0.375
	H-7	14.6	0.373
	H-8	13.1	0.410
	H-9	39.8	0.097
Reflector Irradiation			
Thimbles	V-10	40.2	0.095
	V-11	39.0	0.101
	V-12	12.6	0.424
Core Edge Irradiation			
Thimbles	V-13	1.1	1.080
	V-14	1.1	1.080
In-core Irradiation			
Thimbles (and shroud plates)	V-15	----	1.869
	V-16	----	1.869
Vessel Wall	----	79.3	0.0048

the reactor operations floor, and provides the required height of heavy water over the core for shielding purposes during vessel service operations.

This section of the vessel is provided with a variety of penetrations (see Figure 28), the locations of which are established by service and operational requirements. The penetrations are as follows:

- (a) One primary coolant inlet nozzle.
- (b) One primary coolant outlet nozzle.
- (c) Eight main and auxiliary control rod drive shaft penetrations.

- (d) One emergency coolant inlet nozzle.
- (e) Four irradiation thimble penetrations in the vessel head.
- (f) Five fuel handling penetrations in the vessel head.
- (g) Miscellaneous instrument, siphon break, vent, pressure relief valve, and helium cover gas system penetrations.
- (h) Two nozzles for fuel handling periscopes.
- (i) Four emergency natural circulation core bypass orifice actuator nozzles.

3.6.7 Miscellaneous Internal Structures. There are two core edge irradiation thimbles and two in-core thimbles which enter the vessel through the top head and pass through the upper vessel section to the core region. All thimbles are 2 in. schedule 80, type 6061-T6 aluminum pipe, designed for 275 psig external pressure and 400°F maximum temperature. The operating temperature is maintained below 300°F at all reactor powers by internal D₂O cooling water which supplements the cooling effects of the primary coolant.

The upper shroud directs the primary coolant flow through the upper vessel section to the lower shroud described in Section 3.5.2. It is a 32 in. O.D. cylinder constructed of type 304 stainless steel. It has a wall thickness of 3/4 in. The shroud is maintained at a temperature less than 200°F by the primary coolant. It is flanged to the upper vessel section at the top and its lower transition piece is fitted around the lower aluminum shroud. Figure 30 shows details of the construction.

There are eight 4 in. O.D. main control rod support tubes. These tubes are constructed of type 304 stainless steel. Their primary function is to house the main control rod followers in their passage through the upper vessel section to the shroud transition piece. They also support and guide the auxiliary control rod drive followers.

3.6.8 Reactor Vessel Restraints. To prevent lateral displacement of the vessel as a result of nozzle reactions or seismic forces, a restraint is provided on the upper vessel section. It is designed to permit normal radial and vertical thermal expansions but limits horizontal displacements due to other forces, transferring the loads directly to the biological shield.

Calculations of the reactions on the vessel due to the primary piping, with all thermal expansions and flow reactions, have been made for the sets of conditions shown below:

<u>Condition</u>	Temperature at outlet of:				
	<u>Reactor</u>	<u>Pump-1</u>	<u>Pump-2</u>	<u>HX-1</u>	<u>HX-2</u>
I. Normal Operation 2 loops	134°F	134°F	134°F	120°F	120°F
II. Normal Operation 1 loop	134	134	70	120	70
III. Shutdown flow, Case 1	200	134	134	120	120
IV. Shutdown flow, Case 2	200	70	70	70	70

The resulting forces, F_x , F_y , F_z , and moments M_x , M_y , M_z in a Cartesian coordinate system with "y" up and "x" in the plane of the inlet and outlet nozzles are given in Table XVI. The nozzle reactions are seen to be small, and to offer no difficulty either in vessel stresses or in maintaining the accurate alignment of the vessel relative to the shields.

3.6.9 Vessel Flow Pattern. The flow of heavy water through the reactor vessel is shown in Figure 18. Heavy water enters the 48 in. diameter reactor vessel neck at a velocity of 8-1/4 ft/sec. Contained by the upper shroud and the lower shroud, the heavy water is directed downward through the fuel elements in the core. The velocity of the heavy water past the fuel element plates is 35 ft/sec.

Once past the fuel elements a portion of the heavy water flows downward through the lattice plate. However, to lessen the velocity of water through the lattice plate, some water is allowed to pass directly to the reflector through a series of holes in that portion of the lower shroud which lies below the core.

After passing through the lattice plate the heavy water is diffused by the supportwebs and is turned upward. It then flows through the reflector, that portion of the reactor vessel outside the lower shroud; upward through the annulus formed by the reactor vessel neck and the upper shroud; and out of the vessel into the pump suction line at an exit velocity of 8-1/4 ft/sec.

Table XVI.

Reactions on the Reactor Vessel Inlet and Outlet Nozzles

(Forces are in pounds; moments in inch-pounds.)

<u>Condition</u>	<u>Fx</u>	<u>Fy</u>	<u>Fz</u>	<u>Mx</u>	<u>My</u>	<u>Mz</u>
A. Outlet Nozzle						
I.	-12	+646	-3	+635	+102	+34949
II.	-71	+745	+1	-9874	+27901	+25962
III.	-245	+2470	-35	+8134	+1650	+83423
IV.	-362	+2669	-32	+7518	+1546	+65515
B. Inlet Nozzle						
I.	+195	+65	0	-1	0	+23778
II.	+71	-6	-95	+17201	+14438	+15050
III.	+195	+65	0	-1	0	+23778
IV.	+71	-6	+95	-17201	-14438	+15050

3.7 Shielding

3.7.1 Thermal Shield. The thermal shield is designed to limit the total energy current into the concrete biological shield to a value well below that which might cause excessive thermal stresses in the concrete. The thermal shield also serves as a water-tight pit in which the reactor vessel is placed, and the interior volume of the thermal shield is chosen so that the reactor core will remain immersed in heavy water in the event of a leak in the lower part of the reactor vessel.

The allowable energy current into the concrete depends upon the type of radiation which carries the energy, since the penetration into the concrete and the point of maximum energy dissipation in the concrete depend on the radiation medium. The following list gives, for various radiations, the energy currents into BNL iron-limonite concrete which give a 30°C temperature rise:

8 Mev gamma rays	6.5×10^{11}	Mev/cm ² -sec
3 Mev gamma rays	9.1×10^{11}	"
Fast neutrons (including capture energy)	13.7×10^{11}	"
Thermal neutrons (including capture energy)	13.7×10^{11}	"

A maximum total energy current from all sources of 3×10^{11} Mev/cm²-sec into the concrete has been set as a design basis, and as listed in Section 3.4.14, the calculated current at 40 MW reactor power is about 1×10^{11} Mev/cm²-sec. Most of this is caused by fast neutrons; the gamma flux is attenuated by a factor of 4000 in the thermal shield and amounts to only 1×10^{10} Mev/cm²-sec at the concrete surface.

The thermal shield wall section includes 3 in. of steel plate, 1-1/4 in. of cooling water, and 4-1/2 in. of lead-cadmium alloy (the eutectic, at about 17 wt % cadmium). The construction is shown in Figure 31, and the position of the thermal shield with relation to other components of the system is shown in Figure 25. The thermal shield cooling system is shown in Figures 33 and 34, and is discussed in Section 3.8.10.

The interior of the thermal shield, and the space in the reactor pit above it, up to the pit seal plate (see Figure 25) is filled with CO₂ at atmospheric pressure. This inert gas blanket in the high radiation areas around the reactor vessel prevents both the formation of nitric acid in these spaces, and the formation of argon-41 (as would occur with air). The CO₂ is supplied from gas storage bottles and is vented into the exhaust gas system and passed up the stack.

The cavity seal plate is formed of 1/4 in. thick lead sheet, bonded to the steel walls of the pit, to the reactor vessel, and to all of the penetrations through it. The lead forms a completely tight, flexible seal membrane which is not damaged by radiation. It may be cut open and resealed easily for any maintenance work which is necessary in the lower pit.

3.7.2 Biological Shield. The heavy concrete biological shield is designed to reduce radiation levels outside the shield to values well below tolerance. The calculated radiation levels through the bulk shield are about 0.1 mr/hr gamma

and 1 fast neut./cm²-sec. The same radiation levels are expected around beam holes with shutters closed and blank shield plugs in place. As the blank plugs are replaced with collimators for the beam experiments, and shutters are opened, massive supplementary shielding will be required around the apparatus.

Details of the arrangement of the biological shield are shown in elevation in Figure 25 and in plan in Figures 26 and 27. Note that a thermal barrier gap of 1/2 in. is maintained between the thermal shield vessel and the concrete. This space is filled with CO₂. The average thickness of the biological shield at the core level is 7-1/2 ft. The concrete is faced, on both inside and outside surfaces, with 1 in. thick steel plate, except along the thermal shield, where it is 1/2 in. thick.

The heavy concrete of the biological shield is made of limonite ore, iron punchings, Portland cement, Plastiment, and water. It is a shielding concrete which has been used at BNL in the Graphite Reactor and the Medical Reactor shields. Properties of the concrete are given in Table XVII.

Table XVII.

Properties of BNL Iron-Limonite Concrete

Density	275 lbs/ft ³
Heat Capacity	0.164 Btu/lb-°F
Thermal Conductivity	2.06 Btu/hr-ft-°F
Coefficient of Expansion	13.7 x 10 ⁻⁶ in./in.-°F
Young's Modulus	3.5 x 10 ⁶ psi
Fast Neutron Relaxation Length	7.4 cm
Gamma Absorption Coefficient, 7 Mev	0.118 cm ⁻¹
Gamma Absorption Coefficient, 3 Mev	0.15 cm ⁻¹

3.8 Process Systems

3.8.1 The Primary D₂O System. The primary D₂O system is shown in Figures 33 and 34, and in the isometric drawing of Figure 36. High purity heavy water is circulated at 17,600 gpm around a closed system consisting of the reactor

vessel (DC101), two primary pumps (GA101A and B), two heat exchangers (EA101A and B), and connecting piping. Heat transferred to the D₂O in the reactor vessel is dumped to the H₂O secondary coolant in shell and tube exchangers. At 40 MW reactor power, D₂O enters the reactor at 120°F and leaves at 134°F. It then passes in order through a venturi flowmeter, two pumps and two heat exchangers in parallel, and then back to the reactor vessel at 120°F. Two auxiliary pumps (GA102A and B) provide D₂O flow during shutdowns. The primary system is pressurized by maintaining the helium cover gas pressure in the expansion volume in the reactor vessel at 196 psig. Heavy water purity is maintained by means of a bypass filter and deionizer system to be described in a later section.

The primary system components are arranged in shielded cells in such a manner that maintenance and repair operations can proceed safely with minimum interference with reactor operations. One primary pump and one heat exchanger are arranged in each of two cells, side by side on the equipment floor. The shielding between these cells is adequate to permit access to one cell in which the equipment is shut down while reactor operation continues using the equipment in the other cell. Under these circumstances the reactor is operated at one-half to two-thirds of full power. A third cell houses the reactor effluent piping, flow meter and pump suction shut-off valves. No access to this cell is permitted during reactor operation. Two additional cells are provided for the primary shutdown pumps, thus allowing access to them during reactor operation. The floor plan of the cells is shown in Figure 46.

The primary pumps are of the vertical, in-line, centrifugal type with horizontal suction and discharge, and other characteristics as given in Table XVIII.

Table XVIII.

Main D₂O Pump Characteristics (GA101A and B)

Number of pumps	2
Type	vertical, in-line, centrifugal, direct drive
Fluid temperature	134°F
Fluid specific gravity	1.09
Design capacity	9000 gpm at 60 psi developed head
Maximum fluid pressure	250 psig
Pump material	304 or 316 stainless steel
Speed	1160 rpm
Hydraulic power	317 hp
Motor power	full curve
Electrical supply	2300 V AC

The primary heat exchangers are of the shell and U-tube type, with D₂O on the tube side. H₂O flows in a single pass through the shell side, which is baffled. Data on the heat exchangers are given in Table XIX.

The D₂O side of the heat exchangers is maintained at a higher pressure than the H₂O side to insure that any leakage is away from the primary circuit. Radiation monitors on the H₂O effluent provide a sensitive leak indication. Valves on both primary and secondary sides of the exchangers permit isolation of a leaking unit.

Four 18 in. stainless steel plug valves serve to isolate the pair of primary pump and heat exchanger combinations. The two valves on the pump suction are manually operated from outside the cell by pneumatic means. The two valves on the heat exchanger D₂O effluent lines are motor operated and are controlled from the operations level control room.

Sizes of principal lines range from 16 to 20 in. Schedule 10 and are set to minimize inventory without excessive pressure drop, or danger of cavitation damage. The material of construction is 304 stainless steel. The piping, valves, and flowmeter are assembled by welding. The primary pumps and heat exchangers are flanged. The arrangement of the primary piping and equipment is shown in Figure 36.

Table XIX.

Main D₂O Heat Exchanger Characteristics (EA101A and B)

Number of exchangers	2
Type	Shell and U-tube
Total duty	40 MW
Code requirements	TEMA-R and ASME Section VIII, and Case 1270N
Tube side, per exchanger:	
Fluid	D ₂ O
Flow rate	9000 gpm
Temperature in	134°F
Temperature out	120°F
Design temperature	160°F
Operating pressure	250 psig
Design pressure	300 psig
Maximum allowable pressure drop	10 psi
Fouling resistance	0.0005
Tube material	304 stainless steel
Tube size	3/4 in. O.D., 16 B.W. gauge
Tube sheet material	304 stainless steel
Channel and cover	Carbon steel, 304 stainless clad
Shell side, per exchanger:	
Fluid	H ₂ O
Flow rate	4850 gpm
Temperature in	85°F
Temperature out	119°F
Operating pressure	45 psig
Design pressure	150 psig
Maximum allowable pressure drop	10 psi
Fouling resistance	0.0015
Material	Carbon steel

The plant D₂O inventory is discussed here, even though some of the heavy water is in other process systems. Table XX details the total plant heavy water inventory.

Table XX.

Total D₂O Inventory

I. Primary Circuit

Reactor vessel, core region	1250 gal	
Reactor vessel, upper section	1130	
Main heat exchangers, tubes	2010	
Main heat exchangers, heads	1000	
Pumps, valves, and piping	<u>2380</u>	
	7770	7770 gal

II. Shutdown Cooling Circuit

Heat exchanger shell	90	
Pumps, valves, and piping	<u>140</u>	
	230	8000 gal

III. Cleanup System, Experimental System, and Transfer Lines

Heat exchanger shell	100	
Bed vessels	410	
Filter vessels	90	
Pumps, valves, and piping	<u>400</u>	
	1000	<u>9000</u> gal

3.8.2 Shutdown Cooling System. A separate D₂O loop is installed to cool the reactor core during shutdown periods when the main circulating pumps are not operating. This loop is kept in continuous operation. Since a failure of the main electrical power to the facility is to be expected several times each year, the shutdown cooling loop must be ready at all times to assume the after heat cooling load. In the event of a power failure, the main control rods are dropped and the reactor is shut down. The main circulating pumps coast to a stop, and the reactor flow drops to 500 gpm in about 100 seconds (see Section 3.4.10). The shutdown cooling loop, which can be powered from sources independent of the main electrical supply, assumes the cooling load at 500 gpm flow.

The shutdown cooling loop is shown in Figure 34. The loop components are the pumps, GA102A and B, and the heat exchanger, EA103. The pumps are in parallel, with a common

suction supply from the main D₂O loop downcomer. The pumps discharge through the heat exchanger EA103 into the reactor vessel at a point above the upper shroud flange. The shutdown cooling flow is thus downward through the reactor core.

The shutdown pumps are of stainless steel. They develop 500 gpm at a head of 16 psi. The hydraulic power required is 4.7 hp each and the motors are designed to drive the pumps through the full characteristic curve. The motor speeds are 1160 rpm, and the electrical supply is 250 V DC.

The heat exchanger is a shell and U-tube type with the D₂O on the shell side. Construction is of stainless steel on the D₂O side, and the exchanger is designed for the full primary system pressure and temperature conditions. The H₂O secondary coolant is monitored for leaks in the same manner as the main exchangers, and like them is pressurized so that leakage is into the H₂O side. The H₂O effluent from the EA103 exchanger is drained from the system as part of the normal blow-down flow in the cooling tower system. When the cooling tower system pumps are cut off by a power failure, water flows through EA103 by gravity from the cooling tower storage pool. A sump pump, GA303 (see Figure 35), powered from the emergency electrical supply can be operated to return the cooling water to the cooling towers in the event of a prolonged power outage. The H₂O flow rate through EA103 is 150 gpm.

The shutdown system pumps are driven by 250 V DC motors, as noted above. The electrical supply to them is a three-fold one. In normal operation, the motors are powered from solid-state rectifiers on the main building electrical circuits. Failure of the normal electrical supply leaves the shutdown pump motors drawing from a large storage battery bank. In addition, a propane engine-powered generator set is started automatically by the power failure, and is ready to assume the emergency load in a few seconds. The battery bank life at full emergency load is several hours, so that there is ample time to start the generator set. The propane supply to the generator set, in turn, is from the general BNL propane tank, with a capacity of 18,000 gallons.

3.8.3 D₂O Cleanup System. D₂O purity is maintained by a bypass filter and deionizer system. Normal flow to the system from the main D₂O circulating loop is provided by the

pressure drop across the reactor vessel, as shown in Figure 34. Alternately, D_2O may be processed through the cleanup system from the storage tanks, FA101 and FA102, by either of the transfer pumps, GA104A or B. The experimental D_2O system may also be dumped to FA101 and subsequently processed through the filters and deionizers.

Normally, D_2O flows through one of two sets of filters and deionizers at a rate of 20 gpm. This flow rate is adequate to maintain Na^{24} concentrations in the D_2O low enough so that operators may work safely at the top of the reactor vessel during shutdowns without overexposure to the hard gamma radiation from decay of this isotope. During shutdowns and at such other times as may be required, both sets of filters and deionizers may be used simultaneously at a total flow rate of 40 gpm in order to obtain a more rapid cleanup of ionic solutes and particulate matter.

Each set of filters and deionizers consists in order of a pre-filter (FD101A and B), a cationic deionizer (H101A and B), a mixed bed deionizer (H102A and B), and an after-filter (FD102). All filters have elements of 5 micron nominal pore size. The pre-filter will remove all particulate matter except the hydrated aluminum oxide turbidity which may eventually appear in the system. The after-filter serves mainly to prevent resin fines from the deionizers from entering the primary system.

Based on information obtained from existing reactors, the source of the aluminum oxide turbidity is the fuel element cladding. If turbidity should become a problem in the HFBR, it will be necessary to design and install equipment for its removal. Taps for this purpose are provided on the influent and effluent of the cleanup system.

The deionizer beds are each 18 to 20 in. in diameter, with a 4 ft long section packed with resin. Each bed contains 7 ft³ of nuclear grade resins. Cation beds contain deuterated Amberlite XE-77, a strongly acidic nuclear sulfonic cation exchange resin with at least 95% of the exchange sites originally in the hydrogen form and the balance in the sodium form. Mixed beds contain deuterated Amberlite XE-150 which is an intimate mixture of the above cationic resin (40%) and Amberlite XE-78 (60%), a strongly basic quaternary ammonium anion exchange

resin, supplied originally with at least 80% of the exchange sites in the hydroxide form, and not more than 5% in the chloride form. The deionizer vessel material is 304 stainless steel, and the vessels are designed for the full primary system pressure and temperature conditions (250 psig, 150°F).

The four beds together are estimated to have sufficient capacity to remove the ionic debris resulting from the complete solution of a single fuel element. Each of the two deionizers in a set may be operated separately or in series or parallel. Normally, the removable ionic solutes are largely cationic and the cationic bed is operated in series with the mixed bed so that the bulk of the load is on the cationic bed. Thus radiation levels in the mixed bed are reduced and the life of the radiation sensitive anionic resin therein is extended. Series operation produces dead neutral water which is thought to be adequate from a corrosion viewpoint. If a pD on the acid side should be desired, throttle valves are provided to bypass a portion of the flow around the mixed bed. pD may be determined by measurements on samples taken from the system.

The resin beds are deuterated before use and de-deuterated upon exhaustion. Following de-deuteration, the bed and its container are sent to waste storage. Deuteration is by slow upward displacement of H_2O by D_2O , and de-deuteration is by the reverse procedure.

The filters and deionizer beds are contained in covered pits sunk into the equipment level floor. The connecting piping runs in trenches in the floor which are also covered and shielded. The location of the pit and trench system is shown in Figure 46.

3.8.4 Helium System. Helium is supplied as a cover gas to all of the D_2O free surfaces in the reactor process systems. The storage tanks FA101 and 102 and the surge volume above the D_2O level in the shell of EA102 all carry helium at a few pounds gauge pressure (see Figure 34). The reactor vessel, DC101 has helium cover gas in the expansion volume at the top of the vessel.

The primary system is pressurized by adjusting the helium pressure applied to the reactor vessel expansion volume. The nominal gas pressure for normal operation is 196 psig. The

helium is supplied from a bank of storage cylinders on the equipment level of the building. Appropriate manifolds and pressure reducing valves are used to set and maintain the cover gas pressure in the vessel.

A flow of about 10 scfm of helium is maintained through the reactor vessel expansion volume by the blower GB101. This flow sweeps out D_2 and O_2 generated by radiolytic decomposition or corrosion reactions. Comparison with the cover gas conditions in the Canadian NRX reactor leads to an estimate of a maximum D_2 evolution rate of 4 liter/hr (STP). The helium flow is passed through a recombiner containing platinum supported on alumina pellets where the D_2 is catalytically recombined with O_2 . Since the evolved O_2 in the system tends to be taken up in corrosion reactions, additional O_2 may have to be supplied to the sweep stream before it enters the recombiner. At a circulation rate of 10 scfm, the D_2 concentration in the reactor expansion volume will not exceed 0.03 per cent.

The helium cover gas in the reactor vessel is also expected to contain traces of volatile fission products. The concentration of these will be adjusted to an acceptable level by a continuous purge of the cover gas. The exhaust is through a filter section, FD105, which contains activated charcoal and a Millipore-type particulate filter. The exhaust gases are then passed at essentially atmospheric pressure to the off-gas system, through the main building gas exhaust filters, and are discharged up the reactor stack.

3.8.5 Vents, Drains, and Storage. This section will discuss the vents, drains, and storage facilities connected with the reactor process systems. The exhaust and waste drain systems for the reactor building are discussed in Section 4.3.8.

Helium cover gas from the storage tanks FA101 and 102 and the experimental D_2O heat exchanger EA102 is purged occasionally to remove volatile contaminants. The purge is exhausted into a duct carrying exhaust air from the primary system shielded cells directly to the building exhaust filters, and subsequently to the stack.

The helium purge from the reactor vessel is filtered in FD105 as noted in Section 3.8.4, and then exhausted into the reactor off-gas manifold. The off-gas manifold is maintained

at the lowest pressure in the building. It connects directly to the building exhaust filter system. The off-gas manifold is also the exhaust vent for the various irradiation thimble facilities and the gas jackets on the D₂O pump and control rod mechanism shaft seals.

The thermal shield cooling system cover gas is air, and this is vented into the building exhaust ducts (see Figure 34). The thermal shield cavity blanket gas, which is CO₂, is vented into the primary shielding cell exhaust.

Drains marked "DA" are provided on all D₂O system vessels and equipment. These drains lead into the lower D₂O storage tank, FA101. Transfer of D₂O from FA101 to other parts of the system is accomplished with either of the transfer pumps, GA104A and B. These are low capacity, high head pumps which can operate against the full primary system maximum pressure of 250 psig.

To provide for the collection of D₂O from major primary system leaks, a set of floor drains in the shielded cells lead to the pit in which FA101 is located. This pit is of concrete, set below the equipment floor level, with a concrete slab cover. The pit capacity is 10,000 gallons, or slightly more than the entire plant D₂O inventory. The transfer pumps GA104A and B are set up to take suction in this pit if the need arises. A floor drain from the experimental D₂O process area on the operations floor also connects to this pit.

The D₂O storage tanks are FA101 and FA102. Each tank is of stainless steel, with a capacity of 5000 gallons. FA101 is set below the equipment floor level, where it is a low point for gravity drainage of all of the D₂O systems. The other tank, FA102, is installed above the reactor on the operations floor, where it is a high point with gravity drainage into the reactor vessel (but not against primary system pressure). FA102 acts as a D₂O emergency reservoir for fuel handling operations, and for make-up water to the reactor vessel in the event of boiling, natural circulation afterheat cooling.

3.8.6 Siphon Break. The reactor vessel has been arranged as shown in Figures 18 and 25, with the inlet and outlet nozzles well above the core. Further, the vessel is placed in a water-tight pit. These arrangements have been made to insure that

the core remains covered with heavy water no matter what may happen to the system. Pipe breaks, holes in the vessel, even accidental opening of drain lines on the D₂O system all leave the core covered. The remaining possibility for uncovering the core is that a major breach of the primary loop might occur in such a way as to start a siphon action at the outlet nozzle. Although the connection of the upper shroud to the vessel is not gasketed, the gas leakage through the flange would probably not be large enough to prevent the siphon from forming. Heavy water would then be pulled down through the core, up outside the shroud, and out the outlet nozzle.

To prevent such a siphon from forming, a 4 in. pipe connection is made from the top of the outlet nozzle to the upper vessel (see Figure 34). A valve in this line, VB101, is opened by a low liquid level trip signal. The 4 in. line has a sufficient capacity to break a siphon corresponding to a several thousand gallon per minute leak from the primary system.

3.8.7 Natural Circulation Cooling System. The natural circulation cooling system removes fission product afterheat in the event of loss of all forced flow cooling of the core. Results of calculations on the heat transfer and fluid dynamics of the process are given in Section 3.4.12. The components involved are the emergency bypass orifices and piping shown schematically in Figure 18, the vessel relief valve SV101B, the shutdown heat exchanger EA103, and the "DA" drain system (all shown in Figure 34).

The four bypass orifices are 3 in. diameter openings in the shroud transition plate. Each orifice is closed by a plug which is spring-loaded to open upon loss of forced flow, and is held closed by D₂O pressure when either the main or shutdown pumps are operating. The valve construction and the flow diagram are shown in Figure 37.

D₂O is supplied to the drive piston chamber in the valve at a pressure 5 psi greater than that in the reactor vessel. This higher pressure D₂O is taken from the discharges of the main and shutdown pumps. Failure of all four pumps then equalizes the pressure across the valve, and the spring opens the valve. The driving pressure may also be vented by a pneumatically operated valve in the supply line. This control valve is used to open the bypass plugs in the event of a low liquid level in the reactor vessel. The position of each of the

bypass plugs is indicated by a flowmeter on the high pressure D₂O supply lines. The main driving flow is through the center of the valve and out the four 1/8 in. openings in the plug face. When the plug is seated, this flow drops from about 2.75 gpm per plug to about 0.055 gpm.

Operation of the natural circulation cooling system may be required in either of two general cases; simultaneous failure of all four pumps or a major breach in the primary system. In both cases the fission power is cut off abruptly by scram signals from the pump switch gear, the core pressure difference and flowmeters, or the vessel low liquid level instruments. For the pump failure case, due presumably to electrical supply faults, the primary system is intact and the normal flow coastdown occurs. The flow reversal in the core is speeded by the opening of the bypass orifices and the relief valve SV101B. The bypass orifices open when the pumps cease to develop the necessary sealing pressure, and the relief valve is opened by a trip signal from the opening of the four pump motor electrical contacts. The siphon break valve is also opened by this trip signal.

Opening the bypass orifices during the flow coastdown tends to divert the flow into the reflector through the lower friction path. The opening of the relief valve vents the pressurizing helium, lowers the saturation temperature in the core, and thus speeds the onset of bulk boiling in the core. At 100 seconds after shutdown, the afterheat is about 1 MW. Most of the steam generated is condensed in the water column above the core and the energy absorbed as temperature rise in the D₂O in the vessel. The D₂O level is lowered to the shutdown loop inlet nozzle, and is drained from EA103 by opening the 1-1/2 in. "DA" drain to the storage tank. The relief valve is then closed and steam subsequently generated is condensed in EA103, with the condensate flowing by gravity to the storage tank.

The afterheat power is given as a function of time after shutdown in Figures 16 and 17 (the Perkins and King analysis is used). Integration of the power curve yields the result that the afterheat for the first 1420 seconds after shutdown will bring the reactor vessel D₂O to the atmospheric saturation temperature, 212°F. In this calculation the D₂O level is taken at the outlet nozzle invert rather than higher in the tank (i.e., at the bottom of the outlet nozzle opening in the tank).

The vessel holds 1370 gallons to this point, and the liquid level is 4 ft-3 in. above the top of the transition plate. Further, another 3600 seconds is required (i.e., 83 minutes after shutdown) to boil away enough D_2O to drop the liquid level to 12 in. above the transition plate, assuming no condensation of steam within the vessel. At this time a make-up water flow must be started, since it is inadvisable to allow the liquid level to drop any closer to the transition plate. The afterheat power at 83 minutes is 247 KW, and requires a makeup of 2.4 gpm at 70°F.

The storage tank FA102 is located on the operations level, above the reactor, and D_2O stored there is thus available as make-up water to the reactor by gravity flow. A minimum heel of several hundred gallons is kept in FA102, to provide at least two hours of make-up water flow. If none of the several electrical supplies can be brought into operation by the end of this time, more D_2O can be forced up into FA102 by closing the shutdown exchanger drain valve, shutting off FA101, and opening the transfer line to FA102. Steam pressure then builds up in the reactor vessel and forces D_2O from the main exchangers and piping up into FA102. The head required is 50 ft, or about 25 psig for D_2O .

In the case of a break in the primary system of such nature as to prevent forced flow cooling the sequence of events is much the same. In this case the bypass orifices, the relief valve, and the siphon break valve open on a trip signal from the low liquid level instrument. D_2O which drains from the system flows through the floor drains to the FA101 pit. The transfer pumps are arranged to take suction in this pit when the need arises, so that make-up D_2O is available from this source when the heel in FA102 is exhausted.

3.8.8 Poison Water Supply. A supply of boric acid (or other neutron absorber) solution is maintained in the vessel FA202 (see Figure 34) for use as an ultimate shutdown agency. Sufficient solution is stored to heavily poison the reactor vessel water. The poison tank is normally not connected to the reactor vessel, to prevent accidental discharge of the contents into the D_2O . A hose section supplies the necessary connection. The poison injection may be made against system pressure by pressurizing the tank cover gas space with helium.

Alternately, the poison solution may be injected into the experimental D₂O system, and therein circulated through the four fast flux thimbles which are in and alongside the reactor core. The poison water tank is located on the operations floor of the building.

3.8.9 Experimental D₂O System. The experimental D₂O system is a separate, self-contained process system which provides cooling water for the 16 experimental facilities. It is connected to the primary system only through a transfer line and several "DA" drains. The system circulates 175 gpm to the various thimbles and beam tubes with either of the pumps GA103A or B. The total heat load may be as high as 525 KW, which is transferred to the cooling tower water in the heat exchanger EA102. The D₂O is on the shell side of EA102, with a free surface above the tube bundle and a helium-filled expansion volume in the shell. A deionizer mixed bed with filters on each side keeps the D₂O clean. The bed design is identical to that of the main D₂O cleanup beds (Section 3.8.3).

The system is designed for 250 psig, 150°F conditions, but is normally operated at only a few pounds gauge pressure. The process equipment is located on the operations floor level of the building. Figures 33 and 34 give details of the individual facility circuits.

3.8.10 Thermal Shield Cooling System. A cooling loop filled with deionized H₂O transfers the heat generated in the thermal shield and the beam tube shutters to the cooling tower system. The shield heating rates are given in Section 3.4.14. The total heat load is about 4×10^6 BTU/hr, or 1.15 MW, which is 2.9% of the reactor power. The cooling loop is shown in Figures 33 and 34. The components are the pumps GA201A and B, the heat exchanger EA201, and the expansion tank FA201. The pumps and heat exchanger are on the equipment level of the building and the expansion tank is on the operations level.

The system flow rate is 750 gpm, with a temperature rise of 10.5°F. The large flow rate is dictated by the peak heat flux in the thermal shield, 11,430 BTU/hr-ft², and by the need for a uniform distribution of the flow throughout the cooling annulus in the shield. Either of the pumps GA201A or B will supply the full flow, and only one is operated at a time. Like the shield itself, the cooling system piping and equipment is of carbon steel.

3.8.11 Cooling Tower System. The cooling tower system receives the reactor heat in the various heat exchangers in H₂O, transports it to the cooling towers, and there rejects it to the atmosphere. The system carries the heating loads of the Graphite Reactor Complex, including air conditioning in the Pile Building and in the Hot Laboratory as well as the HFBR load. With the construction of the HFBR, the existing cooling tower system will be extended to five cells and four pumps. The cells have a capacity of 3000 gpm each, and return the water to the tower storage pool at a maximum temperature of 85°F. In normal operation the total system circulation is 13,400 gpm, with 9400 gpm to the HFBR and 4000 gpm to the Graphite Reactor Complex.

The average tower vaporization and wind loss is 500 gpm and 500 gpm total is blown down to the sewers to control the concentration of solids in the water. Of the 500 gpm blow down, 350 gpm is taken directly from the tower storage pool. The remaining 150 gpm is the flow through the secondary side (tube side) of the shutdown heat exchanger EA103. This arrangement insures the cooling water flow to EA103 even when the tower system pumps are off, since the tower storage pool at elevation 124 ft is well above EA103 (elevation 97 ft). The tower storage pool contains 70,000 gallons.

A total average make-up flow of 1000 gpm is supplied from the BNL domestic water supply, which is derived from a number of deep wells on the site. Of this make-up flow, 300 gpm enters the tower system directly while 700 gpm is first passed through the air conditioning system in the HFBR building to take advantage of the 52°F temperature of the domestic supply. The distribution of cooling tower water within the HFBR building is shown in Figure 35.

3.8.12 Thermal Shield Gas Blanket System. The atmosphere within and above the thermal shield, around the reactor vessel is CO₂ at a pressure of -3/8 in. water gauge. The CO₂ region includes the beam tube shutters, the 1/2 in. void space between the thermal and biological shields, and that portion of the reactor pit up to the pit seal plate (see Figure 25). The beam tube plugs are arranged to seal the CO₂ region away from the experimental level air atmosphere.

The CO₂ blanket is provided in the high radiation field areas to prevent the formation of nitric acid, which occurs in damp air in a radiation field, and to eliminate the annoying problem of argon-41 formation in air.

The CO₂ is supplied from gas storage bottles on the equipment level through appropriate manifolds and pressure reducing valves. A slow purge of the CO₂ to the off-gas system serves to keep contaminant vapors to a low level.

3.9 Experimental Facilities

3.9.1 Standard Beam Tubes. There are six standard horizontal beam tubes; five are tangent and one is radial with respect to the core. The radial tube, designated H-2, will be used for fast-chopper, time-of-flight experiments in which neutrons of resonance energy are required. All tubes pass through the aluminum reactor vessel, thermal and biological shields, and terminate in utility boxes on the pile face (see Figures 21 and 26). The tubes are vertically displaced to avoid interference in the reflector region. Their distance from the core varies with the required neutron energy and intended purpose of the facility. Table XXI lists these facilities and their uses. A description of the 3-1/2 in. I.D. beam tubes in the reactor vessel has been given in Section 3.6.3.

That portion of the beam tube which passes through the heavy concrete consists of a shutter, a stepped tube having a close-fitting plug, and a recessed utility box as shown in Figure 32. The shutter operates in a casing which is attached to the thermal shield through an expansion bridge. All joints are watertight and the entire assembly is aligned to within ±1/4 in. along its horizontal axis.

A shield plug is inserted in the stepped tube and is machined so that the total diametric clearance does not exceed 0.090 in. Generous offsets and centering buttons serve to reduce radiation streaming to tolerable levels. A stepped hole within the shield plug is filled with 4 in. diameter collimator blocks or, if the facility is not in use, a heavy concrete filler plug.

Table XXI. Summary of Experimental Facilities

<u>Station</u>	<u>Description</u>	<u>Type of Experiment</u>	<u>Flux</u>	<u>Distance from Core Center</u>	<u>Distance from Exp. Floor</u>	<u>Size</u>
H-1	Beam Tube	Chopper or Diffraction	Resonance	12 in.	3 ft-6 in.	3-1/2 in. I.D.
H-2	"	Fast Chopper-Long Beam	Resonance	12 in.	4 ft-6 in.	"
H-3	"	Slow Chopper	Thermal	18 in.	3 ft-6 in.	"
H-4	"	Diffraction (Dual)	Thermal	18 in.	4 ft-6 in.	"
H-5	"	Diffraction	Thermal	18 in.	3 ft-6 in.	"
H-6	"	Diffraction (Dual)	Thermal	18 in.	4 ft-0 in.	"
H-7	"	Diffraction	Thermal	18 in.	4 ft-6 in.	"
H-8	"	Diffraction	Resonance	16 in.	4 ft-0 in.	"
H-9	"	Cold Neutron	Thermal	23 in.	4 ft-0 in.	12 in. I.D.
V-10	Thimble	Irradiations	Thermal	30 in.	----	*1-7/8 in.
V-11	"	Irradiations	Thermal	30 in.	----	*1-7/8 in.
V-12	"	Irradiations	Thermal	19 in.	----	*1-7/8 in.
V-13	"	Core Edge Irradiation	Fast	10 in.	----	*1.509
V-14	"	Core Edge Irradiation	Fast	10 in.	----	*1.509
V-15	"	In-core Irradiation	Fast	1-1/2 in.	----	*1.509
V-16	"	In-core Irradiation	Fast	1-1/2 in.	----	*1.509

*I.D. of flow divider tube.

The shield plug is an integral part of the containment of the CO₂ atmosphere within the thermal shield. A hemispherical aluminum cap seals the collimator hole at the inner face of the shield plug, while a flange and ring gasket prevent leakage past the outside diameter of the shield plug. Both the cap and flange are designed to withstand transient pressures of 400 psi.

The utility boxes are recessed 12 in. so that service connections can be made within the shield. These connections include D₂O circulation lines, drains, off-gas lines, and 4 in. conduits for access to the upper and lower troughs. Power, cooling, instrument, gas or other services can be led through these conduits.

A gas sweep line opens into the sealed volume between the plug and the liner and this volume is purged before removal of the plug. Handling of active shield plugs or experimental equipment from the beam holes will be done through a shielded cask. Hot plugs are stored in horizontal holes in a concrete vault located on the Experimental Level (see Figure 47).

3.9.2 Standard Beam Tube Shutters. All standard beam tubes are fitted with shutters. These can best be seen in Figure 32. They serve a two-fold purpose; first, to attenuate fission radiations so that minor adjustments or inspection of equipment can be made at the pile face while the reactor is operating, and second, to shield against shutdown radiation while removing a beam plug or setting up an experiment.

The shutters are approximately 7 x 30 in. in cross section and over 6 ft high. The lower portion covers the beam hole in its closed position. It is a boral-lined tank with the 30 in. length divided into 16 in. of water and 12 in. of lead. The balance of the shutter is filled with heavy concrete. Heat is carried from the lower areas by means of a circulating H₂O system. Cooling lines pass through the upper portion of the shutter to flexible hoses before connecting to a header in the reactor pit.

The shutter is raised and lowered by a valve operator mechanism located in a recess at the reactor face. It is connected to the shutter through a shaft. The shaft passes through seals which prevent CO₂ leakage from the reactor pit.

The shutter can be closed in 1-1/2 minutes with this device, and in the event of a power failure, the shutter can be moved by hand with a mechanical overdrive. An off-gas line below the shutter completes the gas purge circuit mentioned in 3.9.1. This line serves as a drain should a leak occur in the shutter cooling system. Means are provided for removing shutters from the pit with the crane on the operating level.

3.9.3 Dual Beam Tubes. In addition to the six standard horizontal beam tubes, there are two beam tubes having enlarged stepped tube and plug sections and wider shutters. They are noted as H-4 and H-6 in Figure 26. These tubes are tangent to the core and are placed in a high thermal neutron field. Except for the larger plugs and shutters the dual beam tubes are identical to the standard beam tubes in construction.

Although the dual beam holes in Figure 26 are shown with one collimator hole in the stepped plug, future stepped plugs will have two slightly diverging collimator openings to accommodate two spectrometers simultaneously. In this application the effective beam size is reduced from the 3 in. diameter of the standard facility to a modified rectangle approximately 1-3/4 by 3 in. The larger openings in the shield will give designers of future experimental apparatus a greater degree of flexibility.

3.9.4 Cold Beam Tube. The cold beam tube, H-9, is shown in Figures 21 and 26. It is a 12 in. I.D. tube which is designed for cold neutron beam experiments. It will be filled with a neutron scattering material such as D₂O or H₂O ice refrigerated to a low temperature.

The end of the H-9 tube is 23 in. from the core center. A closer approach is not considered desirable in view both of gamma heating rates in the thimble wall and the reactivity cost of the thimble. The high gamma heating rate in the cold moderator necessitates the use of a 2 in. thick bismuth shield around the scattering can to reduce the heating rate to about 2.5 kilowatts.

The cold beam tube stepped plug is of the same diameter as the dual beam plug. A shutter has been designed as an aid in plug removal during shutdown.

3.9.5 Thermal Irradiation Thimbles. There are three facilities of this type planned for the HFBR; V-10, V-11, and V-12 (see Figures 25, 26, and Table XXI). Each is constructed of a long straight pipe that fits within the 3-1/2 in. I.D. re-entrant tube in the reactor vessel, and extends upward at approximately 18° from the vertical through the thermal and concrete shields to the reactor top. There it assumes a vertical position in a removable shield plug.

The thimble is cooled by the forced circulation of D₂O in a closed system common to all experimental facilities as noted in Section 3.8.9. The coolant is introduced into the thimble about at its midpoint, flows downward through a flow divider to cool a capsule, and returns through an annulus between the pipe and flow divider. The heat load is expected to be a maximum of 25 KW per thimble. The lower part of the thimble is a 2-1/2 in. Schedule-40 aluminum pipe of 6061-T6 alloy, designed to withstand 275 psi pressure (internal) at 400°F.

The upper end of each thimble is attached to a bedplate which supports the 6-ton handling cask, shown in Figure 38. By use of this sealed cask, samples in aluminum capsules can be charged or retrieved. Station V-12, which passes within 19 in. of the core center, will be fitted with a hydraulic rabbit. In this device the D₂O is both coolant and propellant, with the rabbit functioning as a loose fitting piston. By reversing the flow direction the sample, which rests upon an aluminum screen at the bottom of the flow divider, can be discharged to the handling cask.

3.9.6 Fast Irradiation Thimbles. There are four fast thimbles, as shown in Figure 21. V-15 and V-16 fit within the "egg crate" shroud near the core center while V-13 and V-14 run adjacent to the core just outside the shroud, terminating 11 in. below the horizontal core centerline. All are constructed of 2 in. Schedule 80, 6061-T6 aluminum pipe.

Both core-edge and in-core thimble types consist of three sections. The lower section within the reactor vessel terminates at the positions mentioned above. The thimble lower section is capped with a hemispherical head, and is flanged where it penetrates the vessel top. Here it joins an S-shaped

pipe which is offset to avoid the fuel handling equipment. The third section rises vertically through the top shield to a recessed bedplate identical to those used for V-10, V-11, and V-12.

Each in-core thimble fits into a pressure breakdown block within the egg crate. This block is shaped to duplicate as closely as possible the pressure drop through a fuel element to reduce lateral forces on the egg crate wall and to equalize D_2O flow through all portions of the core. Core-edge thimbles differ from those within the core only in their paths through the vessel--in this case penetrating the shroud transition plate. A spherical seal on the thimble engages a conical seat in the transition plate to prevent excessive leakage at this opening.

At the reactor top, above the shield blocks, the thimbles terminate in a valve lock and recessed bedplate. When the handling cask is sealed and bolted to the bedplate (Figure 38), it becomes an integral water and gas tight unit capable of withstanding not only the pressure of the sample cooling system, but also the 275 psi primary water pressure, which would be brought to bear in the event of a tube rupture within the reactor vessel. Samples may be discharged during reactor operation by use of the valve lock. In this way, one valve is always closed should a rupture of the thimble occur within the vessel.

The following service connections are available to all vertical facilities: D_2O cooling water, drain, off-gas and power leads. A trough in the floor is provided for leads from the cask or sample to recording and monitoring equipment on the operations floor. The cooling water for the fast thimbles is supplied from the experimental D_2O system. The heat generation in the in-core thimbles is calculated at 100 KW maximum each, requiring a 33 gpm flow of D_2O . Heat generation in the core-edge facilities is 50 KW each and 17 gpm of coolant per tube will be circulated.

3.9.7 Safety Aspects of the Experimental Facilities. An important characteristic of the experimental facilities, from the reactor safety standpoint, is the restriction on sample sizes and reactivities. In order to provide the best possible external beams, it has been necessary to restrict the poisoning effects of the irradiation experiments. A reactivity allowance of about 1.0% in k has been made for all

irradiation samples in the reactor at one time, which means that individual samples must be small. Typical samples will be single crystal radiation damage specimens and small (1-2 gram) quantities of radioisotopes. Problems of variable flux peaking around experiment positions due to a wide variety of strong and weak absorbers will be minimized in the HFBR.

The possibility of rupturing one of the fast flux thimbles in the vessel has been recognized and the sample handling cask and all the external process equipment and piping is designed to stand the full primary system pressure. A variety of malfunctions can occur in irradiation facilities; Table XXII lists the more critical ones, the method of detection of the difficulty, and the remedial actions to be taken.

The external beam hole safety questions all center on two problems; the shielding problem when the shutters are raised and the holes opened for experiments, and the problem of maintaining a tight gas and water seal between the internal CO₂ atmosphere and the external air. The shielding problem will differ in its particulars for each experiment, since varying sizes of collimators and hole apertures will be used. As a general rule, each experiment will require the equivalent of 3 to 5 ft of heavy concrete around the first scattering target in the beam. Scattered beams from the target will, in turn, require backstop shielding of 2 to 4 ft of heavy concrete. Each experiment will be monitored for radiation leakage by instruments set up at appropriate locations on the experimental floor.

The cold beam hole, H-9, poses especially severe shielding problems when the experimental equipment is being moved into place and removed. It is expected that a large cave of heavy concrete shielding blocks will be assembled outside the reactor shield for these operations, which will be done only during periods when the reactor is shut down.

The gas seal in the beam holes is provided by the outer shield plug (see Figure 32). This plug has a hemispherical aluminum cap across the beam at the inner end, and is gasketed to the hole liner flange at the outer end. The flanges are all 150 psi class, to ASME Code designs. At the operating temperature of 70°F, the permissible working pressure is 275 psig, and the effective rating is increased by a factor of at least 2 for short time, infrequent pressure surges.

Table XXII.

Safety Aspects of Irradiation Thimbles

A. In-core and core-edge tubes.

<u>Type of Accident</u>	<u>How Indicated?</u>	<u>Counter-Action</u>
1. Thimble rupture within the vessel.	1.a. High D ₂ O level alarm on EA102. b. Pump discharge pressure indicators show rise.	1. Shut down reactor. Replace defective tube.
2. Thimble rupture outside of vessel.	2.a. Excessive flow on flow indicator. b. Blotter paper or "beetle" leak indicators. c. Low level in EA102. d. Low pressure on pump discharge indicators.	2. Valve off thimble. Discharge experiment. Schedule repair at shutdown.
3. Loss of flow.	3.a. High temperature reading. b. No flow on flow indicator.	3. Discharge capsule. Valve off thimble and replace at shutdown.
4. Break in capsule.	4. Shown on conductivity recorders (if sample is soluble), and radioactivity monitors on D ₂ O lines.	4. Valve off thimble. Discharge capsule if possible. Decontaminate thimble or replace at shutdown.

Table XXII. (continued)

Safety Aspects of Irradiation Thimbles

A. In-core and core-edge tubes.

<u>Type of Accident</u>	<u>How Indicated?</u>	<u>Counter-Action</u>
5. Capsule jammed during charging or discharge.		<p>5.a. If stuck in pig, remove pig to hot cell and dismantle.</p> <p>b. If caught in "S"-tube, remotely dismantle and pull up into special shield.</p> <p>c. If stuck in tube within vessel, it can be pulled up as far as possible, cut, and the remainder pulled into a special shield or discharged to the canal, depending upon its shape.</p>

B. Thermal flux tubes.

1. Thimble rupture within re-entrant tube.	1. Excessive flow on indicator, low level in EAl02.	1. Shut down reactor. Discharge sample. Valve off thimble and replace defective tube.
2. Item 2 to 5 similar to above.		2. This facility can be cut and discharged to the canal via the fuel chute.

The aluminum caps have a minimum thickness of 0.100 in., an I.D. of 4-1/4 in., and a bursting pressure above 400 psig. The outer shield plug, and hence the gas seal, may be removed only when the reactor is shut down. At such times the air flow into the hole (and the CO₂ flow out) is controlled by forced exhausts through the utility box and shutter casing 1-1/2 in. drains. In normal use of the holes, the outer shield plug stays in place, and the inner shield plug is replaced by whatever collimating device is to be used in a given experiment.

3.10 Fuel Handling

3.10.1 Procedure. The procedure for discharging spent fuel elements provides for complete separation of the heavy water reactor coolant and the light water in the canal. Each element is lifted out of the core, up through the D₂O and into the pit at the top of the reactor vessel. A slide arrangement guides the fuel element to the top of the discharge chute where the element is released and allowed to drop to the canal (see Figures 39 and 40). During this operation the reactor shielding and a pair of lead shutters protect operating personnel from the high radiation field. An air exhaust system carries any air borne contamination to the off-gas manifold, through the building exhaust filters, and up the stack.

The level of the heavy water in the reactor vessel is raised to the top of the 4-ft diameter neck (see Figure 18) during fuel changing operations to provide additional shielding from the core gamma radiation.

In order to facilitate the grappling of fuel elements, an indexing mechanism is employed to support the lifting head cable. This permits an operator to locate the lifting head above any fuel element in the core, while watching the movements of the cable and lifting head by means of a periscope.

3.10.2 Indexing. The indexing mechanism operates above the pair of lead shutters (shown in Figures 39 and 40) which cover an access port through the top layer of shielding. The mechanism provides a rectilinear motion to cover the area of the core, and also a rotary motion about the central axis of the pick-up head to permit it to be turned, if necessary, to enter the egg-crate cell. The cable take-up reel for the lifting head is enclosed to confine contamination.

3.10.3 Discharge Chute. A 57 ft long discharge chute, built into the biological shield, is located next to the reactor vessel. An element is dropped into the discharge chute and falls directly to the canal. A metal tritium containment shell surrounds the top of the vessel and the top of the discharge chute during the fuel handling operation.

3.10.4 Element Cooling During Fuel Handling. Following a reactor shutdown, a 16-20 hr cooling period will elapse before discharging any fuel element. As shown in Figure 15, the fission product decay heat for the hottest element would then be about 7 KW. Experiments conducted at ORNL indicate that the element, if allowed to stand in air, would eventually melt (32). Normal discharging procedures allow the element to stand in air for less than 30 seconds and the temperature rise in this time would be only a fraction of the maximum. It is, however, necessary to provide air cooling of the element in case of a malfunction of the handling equipment.

This cooling system is designed to keep the fuel element metal temperature below 370°F. The element is lifted into an air bonnet as it comes out of the water and the cooling air is sucked through the element. An emergency cooling system sprays D₂O over the top of the element in the case of an air cooling failure.

3.10.5 Periscopes. The movement of fuel into and out of the reactor can be accomplished with more assurance and greater speed when the operation is one that can be viewed. The fuel handling apparatus is designed, therefore, so that the operator can see the fuel element being transferred throughout the process. In order to accomplish this two periscopes are used; one for the maneuvering of fuel in the reactor vessel, and one to view the element while being moved from vessel to discharge chute. These are shown schematically in Figures 39 and 40.

3.10.6 Mock-up. To assure proper operation of the fuel handling equipment when installed in the reactor, a complete fuel handling mock-up has been built. This consists of a water tank containing an aluminum core and fuel elements, an indexing mechanism supporting the lifting cable and grappling head, the shutter assembly, the tritium containment shell and a

discharge chute. The operation of the mock-up shows that the system will allow the movement of the elements from the core to the discharge chute in about one minute.

3.11 Instrumentation

3.11.1 Nuclear Instrumentation. A block diagram of the nuclear instrumentation is shown in Figure 41. There are eight basic channels which give information on the neutron level of the reactor. All of the neutron sensing elements except the thermopile are installed in instrument tubes which terminate just outside the thermal shield. The thermal shield wall opposite the end of each instrument tube is altered to permit the diffusion of neutrons into the tube. A detail of this neutron window in the thermal shield is shown in Figure 31. An instrument tube is shown in elevation in Figure 25, and the locations of the ten instrument tubes in plan are shown in Figures 26 and 27 (the instrument tubes are numbered I-1 through I-10).

The neutron sensing chamber for each channel is lowered on its connecting cables down the instrument tube from the pile top. The lower end of each instrument tube is connected to the equipment level shield cells by a drain pipe. Air is allowed to flow down the instrument tube to cool the neutron chamber. The level of the full power signal from the neutron chamber is adjusted by pulling it some distance back up the tube.

The instrument channels, 1, 2, and 3 are identical safety channels. Each channel has two ion chambers. One for power level is connected directly to the safety amplifier. The other feeds its signal to a period amplifier, with the output in turn going on to the safety amplifier. The period and power level outputs from the three safety channels are recorded. The period recorder takes the output of each of the three period amplifiers in turn for twenty minutes. The power recorder is cycled in the same fashion on the three power level signals. Comparison of the period and power levels as indicated by each of the three channels in turn is a sensitive method for detecting drifts in calibration or other electronic faults. The period and power level for each channel are indicated on large meters on the face of the amplifier chassis.

For each of the three safety channels, the period and power signals are combined in a safety amplifier to form a single safety signal (see Figure 41). When the safety signal reaches 80% of full scale, the safety amplifier calls for a setback action of the control rods. When the safety signal reaches 100% of full scale, the safety amplifier calls for scram action of the control rods. (In the setback, rods are driven in at normal speed; in scram, the clutches are released and the rods drop rapidly.) The outputs from the three safety channels are combined in logic circuitry and rod action in either scram or setback occurs only when two out of the three channels call for rod motion. At low power levels the safety signal is composed largely of a period input. As the power level rises the safety signal contains more and more of the power component, and at full power a relatively slow reactor period added to the full power signal results in setback or scram.

Instrument channel 4 is a linear power channel, with the compensated ion chamber signal being amplified and recorded. A range switch is provided so that the operator may keep this channel "on scale." A differential power meter circuit is fed in parallel with the power recorder.

Instrument channel 5 is a power channel, with signal taken from a thermopile installed in the cover of the thermal shield so that the sensitive elements extend down into the shield cavity. The thermopile is a relatively slow response power level indicator, but because it involves no electronic amplifiers between the sensing element and the indicating point, it is a valuable power instrument.

Instrument channel 6 is the automatic power level control channel. The neutron level is sensed by one or more compensated ion chambers. The combined signals are amplified in a linear circuit, and the output fed to a controller which operates the drive mechanism of one of the auxiliary control rods. A recorder indicates the signal which is being fed to the controller, and a range switch on the linear amplifier permits the channel to control the reactor in any power range.

Instrument channels 7 and 8 are startup channels. These will be required for the initial startups of the reactor and for any subsequent startups in which all the fuel elements have been replaced. For normal startups, in which half of the

fuel elements have been irradiated in the previous half-cycle and contain fission products, the neutron level due to the γ -n reaction will be high enough to be detected on the power range channels. Each startup channel has a fission counter for neutron detection. Pulses are fed from the counters into linear amplifiers and discriminators, and into log count rate and period circuits.

3.11.2 Process Instrumentation. The various process instruments are shown in Figure 34. In general, the pressure, temperature, and flow rate at various points in all of the reactor process systems are measured and indicated either in the control room or at the measurement point. Critical measurements for the safe operation of the process systems are equipped with alarms and are connected to the reactor safety system. Insofar as possible, separate measuring instruments are used to supply alarms and to supply safety signals to the safety system. As with the nuclear instrumentation, process instruments which are part of the reactor safety system are arranged for two out of three coincidence before a safety action is taken.

In addition to the pressure, temperature, and flow measurements in each reactor process system, the conductivity of D_2O and of the thermal shield cooling water and canal water is measured. Radiation monitors on the secondary effluent lines from the four D_2O heat exchangers serve as sensitive leak indicators.

The critical process variables in the primary system are the system pressure, the liquid level in the reactor vessel, the flow rate, and the pressure drop across the core. The inlet and outlet temperatures across the reactor vessel are measured and are combined with the flow meter signal to give a power level.

3.11.3 Reactor Safety System. A block diagram of the reactor safety system is shown in Figure 42. The safety system collects information from the various nuclear and process instruments, determines when an abnormal condition exists, and automatically takes appropriate action to either shut the reactor down or to warn the operators that something needs attention. Three general classes of signals are received by the safety system: 1) those which require an immediate fast shutdown (scram) of the reactor, 2) those which

require shutdown of the reactor, but at a slower rate (setback), and 3) those which require the attention of the operators but do not warrant an automatic shutdown of the reactor.

1. The fast shutdown, or scram, is accomplished by interrupting the holding current to the eight main control rod drive clutches. This action results in the fast drop of the main rods and an abrupt shutdown of the reactor power. The safety signals which cause a fast shutdown in any configuration are:

- a. scram signals from two out of three of the neutron safety channels,
- b. scram signals from two out of three of the reactor vessel liquid level instruments (low trip),
- c. signals from the primary pump switch gear indicating that both primary pump breakers are open,
- d. signals from the primary flowmeter and core pressure drop instruments indicating abnormal conditions, and
- e. scram action may be called for manually by the operator.

2. The slow shutdown of the reactor is accomplished by driving all of the control rods to their fully inserted positions at normal rod speed (setback). The safety signals which cause such action are:

- a. setback signals from two of the three neutron safety channels,
- b. signals from two of the three primary coolant inlet temperature instruments (high trip),
- c. signals from two out of three of the trip circuits on the shutdown pumps (each shutdown pump breaker and the shutdown loop flowmeter),
- d. signals from two out of three of the primary pressure instruments (low trip),
- e. signals from two out of three of the trip circuits on the thermal shield coolant system (each thermal shield pump motor breaker and the thermal shield coolant flowmeter (low trip), and
- f. setback action may be called for manually by the operator.

In addition, any scram signal starts the control rod drive mechanisms into motion toward the fully inserted positions. Access is provided for additional setback signals from the experimental facilities as particular experiments may require them.

3. Safety signals which result in an alarm in the reactor control room are listed in Figure 42.

3.11.4 Health Physics Instrumentation. The health physics instrumentation includes both fixed station and portable area monitoring equipment, and air activity sampling stations. The fixed station area monitoring system is a ten channel unit. Each channel has a gamma sensitive ion chamber (with calibrating source built in), a meter indicating the local radiation level at the ion chamber and a second meter in the control room, and both visual and audible alarm signals at the ion chamber location. The alarm point is normally set at 10 mr/hr.

Two stations of the fixed monitoring system are located on each level of the reactor building. One station is located in the reactor pit, and one station is located in each of the three shielded cells on the equipment level. As experiments are installed at the various beam holes and thimbles, radiation monitors will be set up at appropriate locations to monitor the activity. The health physics personnel in the building will, in addition, have the normal array of hand radiation monitors.

The reactor building air monitors are of the moving filter paper type in which both particulate and gaseous activity is detected.

4. BUILDING DESIGN

4.1 Design Basis

The design basis for the arrangement of the reactor plant and the type of reactor building is as follows:

1. The plant layout and building must provide for a maximum utility of the experimental facilities of the reactor. The external beam experiments require horizontal beams, a minimum radial clear beam path of 50 ft, a minimum vertical clear height of 18 ft, a floor loading of 4,000 lbs/ft², crane service, and a laboratory for each experiment. The experimental area should be free of radioactive contamination and nonessential traffic. The radiation experiments require free access to the thimble terminations, and an area in which radioactive materials may be handled.
2. The reactor plant must be as compact as possible to minimize the inventory of heavy water.
3. Reactor operations require access through shielding plugs to reactor mechanisms, a fuel storage canal, cranes and elevators to handle heavy objects, and a control room overlooking the principal area of reactor operations.
4. The reactor building must provide a sheltered, air-conditioned environment for the experiments. Secondly, it must be designed and built to contain all possible reactor accidents and excursions.

A variety of plant layouts and buildings were considered in the design. Plant layouts with the reactor machinery above the reactor vessel, or outside the reactor building, were discarded because of greater expense in floor structures and in heavy water inventory, and because these arrangements hampered access to experimental facilities. The reactor machinery was finally located directly beneath the core vessel, and various piping arrangements studied to determine the layout which led to minimum heavy water inventory and acceptably low stresses in the piping system.

The experimental requirements led to a plant layout in which the beam experiments are separated from the irradiation experiments and from reactor operations. The irradiation thimble terminations have all been placed above the reactor on a separate floor level.

Various reactor building shapes have been considered, but it is clear that if the building is to sustain any internal pressure only cylindrical or spherical shapes can be built at reasonable cost. The form chosen here is a hemispherical dome on top of a short cylindrical section, with a heavy reinforced concrete foundation slab. The hemisphere is 175 ft in diameter and rises about 100 ft above grade. The structure is divided into three basic levels. A central column of heavy concrete forms the biological shield for the reactor and is also a support member for the various floors.

4.2 Building Layout

The plot plan for the HFBR building and auxiliary structures is shown in Figure 43. Elevations of the reactor building are shown in Figures 44 and 45. Floor plans for the various levels of the building are shown in Figures 46 through 49, and these levels are discussed in the individual sections which follow.

4.2.1 Equipment Level. The general plan of the equipment level is shown in Figure 46. The central section of the equipment area is occupied by the reactor machinery in three shielded cells.

The fuel storage canal extends to the east from the central shielded structure. The fuel discharge chute from the top of the reactor vessel passes down through the concrete shielding and into the canal. The canal is 20 ft deep for most of its length, with a 30-ft deep pit under the fuel discharge chute. The gamma facility is simply an extension of the canal in which irradiated fuel elements may be arranged to provide a gamma source. At the east side of the building a truck airlock provides the necessary entry for handling the shipping casks for spent fuel.

Three rooms along the south wall of the equipment level are partitioned off from the rest of the area and are not part of the contained volume within the reactor building.

These are the transformer room, the filter room, and the emergency generator room. Each of these areas has access from outside the building and is ventilated directly from the outside atmosphere.

The entrance wing on the south side of the building is shown in Figure 46. The entrance wing is not part of the contained volume of the building. It is connected to the building through an airlock.

4.2.2 Experimental Area. The experimental area is shown in Figure 47, and the balcony which is a part of this area is shown in Figure 48. The reactor structure occupies the center of the experimental area, with a large clear space around it for the external beam experiments. Laboratories are located along the wall of the building. On the north side of the experimental area there is a service wing with truck airlock entry to the floor. A machine shop and storeroom are also located on the experimental level.

A traveling-beam radial crane of 20-ton capacity services the experimental area from the pile face to the line of columns.

The balcony level, Figure 48, is located above the laboratories. Washrooms are located there. The remainder of the balcony will be used in the future for office space.

4.2.3 Operations Area. The operations area, on the top floor of the building, is shown in Figure 49. The reactor shielding structure rises 7-1/2 ft above the floor level in the center of the area. The reactor control room, offices for operating personnel, and a washroom are located on the east side of the area. A process room for the experimental D₂O system and other equipment is located southwest of the reactor top. A 20-ton traveling-beam crane operates over the central part of the floor area to service the reactor top and adjacent experimental equipment.

4.3 Utilities

4.3.1 Electrical Power. The HFBR building is supplied power at 2.4 Kv. via an underground conduit system from the 2.4 Kv. switchgear in the Fan House (Bldg. 704). The power feeder terminates in 2.4 Kv. switchgear located on the

equipment level of the building. From the 2.4 Kv. switchgear, a separate feeder is run to each of the two 1000 Kva, three phase, 2.4 Kv. to 480 volt transformers. Feeder cables in conduits run from the secondary terminals of each transformer to the incoming line units of the 480 volt switchgear. The 480 volt switchgear is arranged with a bus tie air circuit breaker to automatically transfer the 480 volt load to one transformer in the event of loss of the other transformer or its feeder. The 480 volt switchgear feeders supply motor control centers and 480 volt panels, which, in turn, feed the motors, lighting panels and experimental station panels.

The total building load is 2,750 Kva, of which 750 Kva is used at 2.4 Kv. by the primary pump motors. The experimental panels are supplied 1000 Kva at 480, 208, and 120 volts and the reactor machinery, lighting, and building services use the remaining 1000 Kva at 480, 208, and 120 volts.

Power into the building is derived from a 6,500 Kva transformer which reduces 13.8 Kv power from the Laboratory substation to 2.4 Kv. This transformer and an identical unit which serves the Graphite Reactor, are located at the Fan House. A normally open bus tie breaker between the 2.4 Kv secondaries permits power to be supplied to both reactors from either transformer in the event of difficulties.

Power to the BNL substation is supplied by the Long Island Lighting Company from a generating station in Port Jefferson, New York. The electrical supply to the Laboratory has been increasingly reliable in recent years, although short-term outages continue to occur from time to time.

4.3.2 Emergency Electrical Power. Electrical power for emergencies is supplied from two sources. The first of these is a propane engine-driven generator set of 125 KW continuous rating at 440 volts, three phase, 60 cycles. The connected load on this generator is 62.5 KW, from the following items:

air compressor	18 KW
building exhaust fan	12
emergency instrument power	5
emergency lighting	15
shutdown rectifier	<u>12.5</u>
Total	62.5 KW

The propane engine generator set is started automatically by a failure of the normal power supply. Five to ten seconds are required in normal circumstances for the generator to come up to speed and to assume the full emergency load.

The second source of emergency electrical power is a battery bank of 160 KW hours total capacity at 250 volts d.c. The battery bank is located in the Fan House and is kept charged by a motor generator set at that location. The battery bank is shared with the Graphite Reactor. The connected loads on the battery bank will be as follows:

Graphite Reactor lighting and instrument load	23.5 KW
HFBR emergency lights	15
HFBR shutdown pump	<u>7.5</u>
Total	46.0 KW

The total battery bank life at full emergency load is 3.5 hrs. This life may be extended somewhat by reducing the emergency lighting load in both the Graphite Reactor and the HFBR after the buildings are evacuated by all but reactor operations personnel. This procedure would be carried out as it became clear after a failure of the normal power that it was going to take some time to start the propane engine generator set. Experimenters and non-essential operating personnel would leave the building and the lighting load would be reduced substantially. The possible reduction by these means is 20 KW, which would about double the battery life.

4.3.3 Communications Systems. There are two general communications systems in the HFBR complex, together with a paging system. The first communications system is the normal Laboratory telephone service. It is expected that telephones will be installed in all of the offices, laboratories, locker rooms, the reactor control room, and in various other locations on the several floors of the building.

The second communications system is an internal system for the use of the Reactor Operations Division. It provides closed circuit communication between the reactor control room, the pile top, the reactor pit, the shielded equipment cells on the Equipment Level, the canal area, and the Experimental Area around the biological shield. The internal system allows operators engaged in servicing the reactor to communicate with each other without tying up the normal telephone system.

The paging system has microphones in the control room and the south wing entrance lobby, and loud speakers in all areas of the building.

4.3.4 Fire Alarms. The building fire alarm system has both manual and automatic alarm stations. Combination rate of temperature rise and fixed high temperature sensing elements transmit a fire alarm signal to the Brookhaven Fire House and sound alarm bells throughout the building. The four floors of the building are each divided into two alarm zones, and additional zones are provided for the south wing lobby and for the north service wing. There will be ten manual fire alarm stations located throughout the building.

4.3.5 Nuclear Incident Alarm. Two fixed station radiation monitors on opposite sides of the experimental level are connected to the nuclear incident alarm. These monitors have an alarm trip setting of 50 R/hr. In the event of a major release of radioactive material to the reactor building atmosphere, the nuclear incident alarm signal is transmitted to the Laboratory Police Control Center. At the same time, the building air intake and exhaust valves are automatically closed.

4.3.6 Steam, Heat, and Air Conditioning. The building is heated by steam heated hot water. The flow diagram for the steam and heating system is shown in Figure 35. The Laboratory steam supply comes into the building from the Laboratory steam system at 90 psig and a maximum rate of 8100 lbs/hr. It is reduced to 15 psig for building use, hot water heating, air conditioning, and miscellaneous uses.

The building air conditioning refrigeration unit is of the steam absorption type and requires about 5100 lbs/hr of steam. The air conditioning is provided by two large units in each of the major areas of the building. Fresh air is brought into two of these units, on the balcony level, filtered, dehumidified, and cooled and is then circulated in the separate building areas. The chilled water supplies to the air conditioning units are from the steam absorption refrigeration unit, and from the 54°F well water supply.

4.3.7 Water Supply. Water is supplied to the HFBR complex from the BNL domestic water system. This system consists of a number of deep wells and an elevated storage tank. The well water is filtered and chlorinated to meet drinking water standards. The temperature of the domestic water supplied to the building will be between 52 and 54°F. The expected flow rate to the HFBR is about 1500 gpm. Of this, 700 gpm is routed through the air conditioning units to utilize the low temperature. Three hundred gpm is added directly to the cooling tower storage pool. Fifty gpm flows through the canal heat exchanger. Two hundred and fifty gpm are assigned respectively to operations intermittent loads and experimental intermittent loads.

4.3.8 Drains and Sewers. There are five general drain and sewer systems installed in the HFBR complex. The first is the "DA" drain system for the D₂O process systems. These drains have been discussed in Section 3.8.5. The "DA" system is a closed, stainless steel piping system to drain D₂O process equipment into the D₂O storage tanks.

The second drain system is the D₂O floor drain system. All of the rooms and areas in which D₂O equipment is located are provided with curbs or trenches so that any D₂O spillage will be retained in the immediate area. Floor drains in these areas will carry off the spillage to the collection pit in which FA-101 is located. The D₂O may subsequently be pumped out of this pit and back into the process systems through the cleanup beds. The D₂O floor drains are not trapped, but the dump pit atmosphere is vented to the building exhaust ducts, and a flow of air into the drains is maintained by the pressure differential thus generated.

The third drain system is the radioactive, aqueous waste "D" system. The "D" waste drains collect radioactive fluids, excluding D₂O, from the various areas of the building and carry them to a collection sump in the basement of the building. A pump, GA-304, transfers the "D" waste to the collection tanks in the Hot Laboratory where the Laboratory "D" waste is held for disposal.

The fourth drain system is the "F" waste system. "F" waste liquids are those which may possibly contain radioactive materials. The "F" waste system drains into a storage

vessel, FA-305, in the basement of the building. The tank is sampled periodically to determine whether or not the level of contamination requires special handling. If there are radioactive materials present, the "F" waste is dumped to the "D" waste sump and is pumped to the Hot Laboratory. If the "F" waste tank does not contain any appreciable amount of radioactivity, it is drained to the sanitary sewer system. "F" waste basins, showers, and floor drains are located throughout the reactor building. The "F" waste collection tank and associated piping are shown in Figure 35.

The fifth drain system is the building sanitary sewer system. This includes the normal drains from washrooms, drinking fountains, and other nonradioactive liquid waste sources. The sanitary drains empty into the Laboratory sewer system. The sewerage is treated, analyzed for possible radioactive material content, and is eventually drained into the Peconic River to the east of the Laboratory.

4.3.9 Air Supply System. The building compressed air supply system is shown in Figure 35. Two compressors of 95 scfm capacity each at 100 psig supply air to the breathing air system, the instrument air system, and the general building compressed air mains. The compressors are driven by electrical motors, and one of these is alternately powered from the propane engine generator set so that compressed air continues to be available for all purposes during a failure of the normal electrical power supply.

4.4 Building Atmosphere Control and Exhaust

4.4.1 General Air Control and Distribution. The general basis for the atmosphere control in the HFBR complex is that air should move from radioactively clean areas toward less clean areas. Further, air should tend to leak into the building rather than out, and all of the building air exhaust should pass through a filter system and be discharged up the high stack associated with the facility. This general scheme is shown in the drawing of Figure 50. Fresh air is brought into the building through check valves and supplied to the various building areas. Since the experimental area is designed to be a relatively clean area, it is maintained at the highest pressure within the building. The experimental

area design pressure is $-1/8$ in. H_2O , so that it is still below atmospheric pressure, and any leakage through openings or cracks in the building must be inward. The operations area and equipment area are maintained at a pressure slightly lower than that of the experimental area, $-1/4$ in. H_2O . Although the stairways which connect the various levels of the building, and also the elevator shafts, have isolating doors, a substantial air flow from the higher pressure area to the lower pressure areas is to be expected.

The volume within the shield is called the primary containment space, and is the most likely area for contamination by radioactive material. It is maintained, therefore, at a lower pressure than the operations area and the basement, $-3/8$ in. H_2O . Air will leak into the primary containment space at a rate about 200 cfm through cracks in the top shielding plugs over the reactor pit, and also through dampers into the shielded cells in the basement.

Exhaust air from all building areas is passed through a three-layer filter bank before it leaves the HFBR building. The filter bank is located in a room in the equipment area as shown in Figure 46.

Two exhaust fans in parallel extract 10,000 cfm of air from the building. This flow is passed through an underground duct to the base of the Graphite Reactor stack. The stack, which handles the large volume of air passing through the Graphite Reactor, is 27 ft I.D. at the base, 17 ft I.D. at the top, and rises to an elevation 350 ft above the general level of the Laboratory. The stack exhaust flow is monitored for radioactivity by suitable instruments in the exhaust ducts.

The various air pressures within the reactor building are adjusted by dampers on the inlet air flow and on the exhaust flow. The building air pressure control is maintained in a failure of the normal electrical supply by powering one of the exhaust fans from the propane engine generator set. Further, the very high stack of the Graphite Reactor has an experimentally determined minimum draft at its base of -0.16 in. H_2O (33). In the event of a loss of all power to run the exhaust fans in the HFBR, the inlet air dampers for the building would close and the building would be maintained at a negative pressure of $1/16$ to $1/8$ in. H_2O .

In the event of a major release of radioactive material to the building atmosphere, a trip signal from the nuclear incident monitors causes butterfly valves on the building air intake and exhaust lines to close within one minute.

4.4.2 Exhaust Air Filters. As noted above, all of the exhaust air from the reactor building passes through the filter bank. There are three sections in series in the filter; a roughing filter, an iodine filter, and an absolute filter. The roughing filter is a deep-bed type with a spun glass medium. There are 12 units in the roughing filter, each 27 in. x 27 in. x 36 in. deep, to give a total face dimension of 6ft-9 in. by 9 ft-0in., and a total face area of 60.7 ft². At a flow from the building of 10,000 cfm, the roughing filter carries 165 cfm/ft², at a velocity of 2.75 ft/sec. The pressure drop through the roughing filter is .25 in. H₂O clean and .50 in. H₂O fouled.

The iodine filter is the Harvard-design, silver-plated copper mesh type. The face area is 6 ft-9 in. x 9 ft, or 60.7 ft². The bed depth is 4 in., with a mesh density of 30 lbs/ft³ and a surface area of 27.5 ft²/lb. The velocity through the iodine filter is 2.75 ft/sec at full flow. The iodine filter pressure drop is .125 in. H₂O clean and .25 in. H₂O fouled.

The absolute filter is rated for 99.95% removal of 0.3 micron or larger particles. It is composed of 12 units, each 2 ft x 2 ft x 5-7/8 in. deep, for a total face area of 48 ft². At full flow, the absolute filter carries 210 cfm/ft². The pressure drop through the absolute filter is 1.0 in. H₂O clean and 2.0 in. H₂O fouled.

The filter bank is contained within a sheet metal envelope which, in turn, is located in the filter room on the Equipment Level of the building. The filter room lies outside the contained volume of the building. Access to the filter room is normally from the outside, although gasketed doors may be opened to reach it from the inside.

The filter banks are provided with blast closure devices on the influent side to protect them from large velocity pressures. Tests have shown that visible structural damage to the absolute filters will occur between 2.5 and 3.0 psig overpressure in the building (34). There are three closure valves located in the containment wall between the filter bank and the main exhaust air duct which empties into it.

The valves, which are shown in Figure 52, are similar to Civil Defense protective shelter air intake valves. A closure disk is mounted on a shaft set centrally within a circular steel pipe. The disk is held away from the entrance a sufficient distance to allow normal air flow around the annular space and into the filter bank. This position is set by a spring with enough force to hold the disk against normal air flow pressure without vibration. A sudden increase of flow pressure will move the disk against the spring and reduce the flow through the pipe; a large increase will close it entirely and engage a spring latch to hold it closed. The closure valve will be set to trip at approximately 8 in. H₂O, or .28 psig. The latch can be opened from outside the building by manual operation of a cable release.

4.4.3 Shielded Cell and Off-gas Exhausts. The three building exhaust gas systems are shown in Figure 51. The first of these is the general building air exhaust system. The total volume of this exhaust is 10,000 cfm. It is carried through the building exhaust gas filters and passed up the Graphite Reactor stack as discussed in previous sections.

The second system is that which exhausts the primary containment spaces. These include the upper reactor pit, the three shielded equipment cells beneath the reactor, and the trenches and pits in the D₂O cleanup system area. A general pressure level of -3/8 in. H₂O is maintained in these spaces by exhausting them into the main building exhaust duct in a venturi orifice on the main stream. The primary containment space is maintained at a lower pressure than other building areas to insure an in-leakage of air rather than the reverse.

The third system is the radioactive off-gas manifold system. The off-gas manifold is maintained at the lowest pressure in the building, -1/2 in. H₂O, by attaching it to the minimum pressure point in the venturi orifice on the main exhaust gas duct. The off-gas manifold takes suction on the gas jackets on the control rod drive mechanical shaft seals, and on the various experimental irradiation thimbles. The purged gas from the reactor vessel is also dumped into the off-gas manifold after it passes through the filter FD105.

The off-gas manifold also handles the cooling air flow during fuel handling. The fuel handling bonnet is connected by a flexible hose to the off-gas manifold for these activities,

and a blower provides the necessary pressure drop. The blower, GB-302, is rated for 300 cfm at 25 in. H₂O. This is an adequate flow to cool the fuel element when it has been withdrawn from the reactor vessel and also to exhaust the tritium vapor hood in the reactor pit during fuel handling.

Both the primary containment space and off-gas manifold exhaust ducts join the main building exhaust duct before it reaches the exhaust air valves and the building filter banks. All gas from these systems, therefore, passes through the filter bank before it is exhausted to the reactor stack.

4.5 Building Structure Containment Features

4.5.1 General. The reactor building shell is designed to withstand hurricane force winds, snow loads, and horizontal accelerations of 0.1 g corresponding to Intensity VIII earthquakes. The building shell is in addition designed to contain an internal gas pressure of 2 psig, with a leakage in 24 hours of 1% of contained volume. The building has a net volume of about 1.7×10^6 ft³, so that the energy release required to raise the building air pressure to 2 psig is 1800 MWS. All openings in the building shell are arranged so that they close automatically in the event of a sudden overpressure within the building, or in the event of a major release of radioactive material.

4.5.2 Steel Dome. The dome of the reactor building is best shown in the elevations of Figures 44 and 45. The dome is hemispherical, 176 ft in diameter, with its base at the experimental floor level. The dome is constructed of carbon steel. An internal frame structure of steel members supports a 3/8 in. thick steel sheet covering. The dome is insulated to reduce the load on the air conditioning and heating systems.

The floor structures within the building, in particular the operations floor and balcony floors, are arranged so that they transmit no bearing loads to the steel shell. The operations floor is supported by the central concrete column of the reactor shield and by a set of peripheral columns at a mean radius of about 65 ft. The operations floor is cantilevered outward from this row of columns and stops short of the steel building shell. A flexible membrane sealer is

extended out to the building shell for air control purposes. The balcony level is supported by the row of heavy columns at 65 ft radius and by a second set of columns along the building wall.

Two designs for the cylindrical building wall beneath the hemisphere are being considered. In the first design the steel shell of the hemisphere extends downward to the foundation level. The shell is secured to the foundation structure by a continuous base ring which engages the foundation. A concrete bearing wall is poured inside the steel cylindrical section. In the second design the steel shell stops at the experimental floor level, and the cylindrical wall is of concrete. The steel shell is tied to the lower wall and foundation in this case by anchor rods. In both designs effective sealing against air leakage is achieved by a deep contact area between the steel and concrete, with a mastic compound filling the interface region.

4.5.3 Foundation Structure. The foundation structure of the building is a thick, heavily reinforced concrete slab. The general soil formation beneath the building is well suited to the heavy load, being a firmly bedded sand with some gravel which extends uniformly to great depths. The soil is conservatively rated at 4 tons/ft² bearing capacity, and the mat foundation will transmit the building loadings with negligible settlement or distortion.

Those portions of the concrete foundations which form part of the gastight shell of the building are coated on the inside surface with appropriate sealing compounds to minimize leakage through the concrete. The various pipes for drains and sewers which must pass through the concrete foundation are fitted with long leakage path collars and are packed with mastic to seal them to the surrounding concrete. They are all internally liquid trapped to hold 2 psig. The canal structure, which is part of the foundation slab, is again a thick-walled, heavily reinforced concrete monolithic structure.

4.5.4 Airlocks and Other Openings. The use of the HFBR as an experimental facility requires much greater access through the containment shell than is the case in power reactors. The experimental equipment which will be used at

the HFBR will be large and heavy. Very large truck airlocks are, therefore, necessary. There are two such airlocks provided in the building, one opening directly onto the experimental area floor, and the other in the basement of the building for easy access to the canal. The truck airlock structures are of reinforced concrete, coated on the inside with a sealing compound. The locks are 65 ft long by 14 ft wide and the doors are 12 ft wide by 14 ft high. The doors are welded steel structures hinged to swing as a single panel. The doors seat upon soft rubber gaskets and are manually dogged closed. The doors are arranged so that an overpressure within the building forces them more tightly upon the gaskets.

A combination freight and personnel airlock opens upon the experimental floor from the north service wing receiving area. This airlock is 9 by 18 ft in size, with 6 ft wide by 8 ft high doors. The doors of this lock are of the flush sill, inflatable gasket type.

A personnel airlock connects the south wing entrance lobby with the interior of the containment shell. This airlock also has flush sill, inflatable gasket doors.

In addition to the various airlock entries to the containment shell, there are in the equipment area several doors to rooms which are outside the contained area. These doors are marine-type bulkhead doors, manually dogged down on soft rubber gaskets. They are provided for convenient access from the inside of the building to the filter room, the emergency generator room, and the transformer room. These doors are all normally closed while the reactor is operating.

The remaining openings in the containment shell of the building are the fresh air inlet ducts and an exhaust air outlet duct. These openings are equipped with valves which close automatically in the event of an overpressure within the building, or in the event of a major release of radioactivity, as has been discussed in Section 4.4 above.

A variety of pipes and electrical cables must be passed through the containment shell of the building. These are sealed to the containment shell in the conventional manner with plastic sealing compounds.

4.5.5 Primary System Containment. In view of the large internal volume needed in the reactor building for experiments, it is not economical to construct the building shell to endure the heavy transient loadings which might result from reactor accidents. The primary system shielding is, therefore, constructed to absorb and confine all violent excursions which might occur, leaving to the building shell the sole containment function of low pressure gas barrier. In particular, care is taken in the design to insure that missiles would not be formed by loose plugs or spalling concrete from the primary shielding, and that the building shell would not be loaded by rigid connection to the operations level floor.

5. PLANT OPERATION

5.1 Administration

5.1.1 Organization. The operation of the High Flux Beam Research Reactor is the responsibility of the Reactor Division of Brookhaven National Laboratory. The Reactor Division also operates the Graphite Research Reactor and the Medical Research Reactor. The organization of the Laboratory for operation of the reactors is shown in Table XXIII.

The functioning of the Reactor Division is such as to provide a maximum of service to the research users of the reactors, and at the same time to provide such limitations to reactor operating conditions as are necessary to protect the BNL staff, the general public, and the investment of the Government in the facilities.

5.1.2 Standard Operating Procedures. At an appropriate time before the operation of the HFBR commences, a Manual of Standard Operating Procedures will be prepared and reviewed, and filed with the AEC. Each operator and supervisor will be provided with a personal copy of this manual. The manual is kept up to date by a reactor engineer to whom this responsibility is assigned. The Manual of Standard Operating Procedures is supplemented by a temporary Operating Procedures Log Book in which instructions will be written on approved changes in operating procedures, either temporary or permanent, until the Manual of Standard Operating Procedures can be brought up to date.

It is the purpose of these Standard Operating Procedures to describe in detail the manner in which each phase of reactor operation is conducted. Although there may be many variations on these prescribed methods which would achieve the same results, such variations are not permitted.

The engineer in charge of keeping the Manual of Standard Operating Procedures up to date also has the responsibility of explaining the changes to the shift supervisors, who, in turn, make explanations to the operators. It is the responsibility of the engineer in charge to see that each operator and supervisor inserts changed sections in his personal copy of the manual, and removes outdated pages.

The approval of the Head of the Reactor Division and of the Reactor and Critical Experiment Safety Committee is required for any changes in the Manual of Standard Operating Procedures.

Table XXIII

Organization of BNL for Operation of the Research Reactors

Laboratory Director

Advisory Committees:

Reactor and Critical Experiments
Safety

Radiation Safety

Reactor Division

Graphite Reactor

Operations

Special Services

Medical Research Reactor

High Flux Beam Reactor

Instrument Maintenance

Mechanical Maintenance

Design

Isotopes and Special Materials

Instrumentation and Health Physics Division

Health Physics

Radiation Survey

Non-Radiation Safety

Meteorology

5.1.3 Training of Operators and Supervisors. The comparatively small turnover of operators and supervisors precludes the necessity of classwork training for new employees. Prospective operators are given a mechanical aptitude test on which the successful candidates are required to make a score which is better than 60% of college first-year students taking the same test. (Test of Mechanical Comprehension, Form BB by George K. Bennett and Dinah E. Fry.) Prospective reactor operators are given a pre-employment physical examination. Prospective reactor operators are hired on a six-month probationary period. The long probationary period is required for reactor operators because they are first trained and examined by the Health Physics organization. Only after having successfully completed the Health Physics training do they commence training on the reactor. It is felt necessary to observe the prospective operator's work on the reactor for several months during the probationary period. The testing of prospective reactor operators and their extended probationary period are intended, in part, to eliminate candidates except those who show reasonable probability of ultimate advancement to 'A' operator status.

The Health Physics phase of the operator's training lasts a minimum of two weeks. The Senior Reactor Health Physicist examines the operator at the completion of this period. The trainee must be able to survey for himself the normal radiation hazards that he is likely to encounter in his work as a reactor operator, compute working times, and be familiar with protective clothing, instrumentation and other safety devices. As a part of this training program he observes other phases of the Health Physics work including waste disposal, personnel monitoring, instrument calibration and urine analysis.

This phase of the operator's training should provide him with confidence that the Laboratory is taking adequate steps to assure his safety and, at the same time, the Laboratory gains confidence that the reactor operator trainee is competent to recognize hazardous conditions and take corrective or protective action.

The remainder of the probationary period is spent in on-the-job training in which the operator is assigned as an "extra" man on a reactor operating shift crew. At the end

of a six-month period the progress of the operator is examined by the Operations Group Leader or the engineer in charge of procedure. At this point the operator's potential ability to achieve "A" operator status is again evaluated. In the event the examiner determines that the probability of ultimate achievement of "A" operator status is low, the trainee may be transferred or terminated without recourse to the grievance procedure. If his training is continued, he is considered a regular operator in training.

At the end of twelve months of employment the operator is examined to determine his fitness for promotion to "B" operator. If the Reactor Operations Group Leader determines that he has successfully passed this examination he will be promoted. If not, he will continue his trainee status for an additional six months at the end of which he is re-examined. If he fails to pass the examination for "B" operator the second time he is terminated. (This has never occurred in practice.)

The "B" operator examination is primarily a practical one in which the operator demonstrates his ability to perform specified portions of the reactor operator's work according to the Standard Operating Procedure.

It should be noted that successful completion of the "B" operator examination is considered as full qualification of the individual to perform all the required functions of a reactor operator. In addition to providing a check of the individual's qualification, the "B" operator examination also provides another opportunity for screening out those individuals exhibiting low potential ability for future advancement.

When the operator has been classified as operator "B" for 18 months, he is examined for his fitness for promotion to "A" operator. In the event of failure, he is entitled to re-examination after a six-month period.

The "A" operator's examination is a comprehensive written examination requiring from four to eight hours to complete. It covers a wide variety of reactor subjects including phases of reactor technology. The subject matter of this examination is specified in advance, but the questions are changed for each examination.

Achievement of "A" operator status reflects possession of a degree of background knowledge and reliability beyond that routinely required of a reactor operator. An "A" operator might be expected to perform certain operations independently, or with less supervision than that anticipated in the case of a "B" operator.

The program of training operators endeavors as its goal to provide the reactors with all "A" status operators. There is, therefore, no specified proportion of "A" operators to "B" operators and trainees in the Table of Organization.

Some of the shift supervisors are former reactor operators who have had a number of years of experience, demonstrated the ability to make the technical decisions required by the job and who possess the necessary leadership qualities. Other shift supervisors are engineering school graduates who are trained as shift supervisors. They perform in this position for two or three years as part of an over-all training program which is designed to provide a source of competent people for other positions. Only qualified operators of "A" and "B" grade and qualified shift supervisors are allowed to operate the research reactors. (Training of others at reactor consoles is done under close supervision by qualified personnel.)

Before being allowed to operate the HFBR, reactor operators will be trained and examined to insure their familiarity with the HFBR system and the HFBR Standard Operating Procedures.

5.2 Normal Operation

5.2.1 Startup and Power Operation. The normal startup configuration of the reactor will include the following:

- a. The core loading is 28 fuel elements, of which the outer 14 are repositioned from the previous half-cycle and the inner 14 are fresh fuel elements.
- b. Both main and auxiliary control rods are fully inserted.

- c. The D₂O level is established in the reactor vessel at elevation 132 ft-4 in., or 3 ft-6 in. over the centerline of the inlet nozzle.
- d. The vessel has been sealed and leak tested, helium circulation has been established, and the helium gas pressure set to 196 psig.
- e. Both primary pumps are running and the various flow-meters and pressure difference indicators show that normal coolant flow has been established.
- f. One primary shutdown pump is running and appropriate flow is indicated in the shutdown loop.
- g. D₂O coolant is circulating in the experimental systems.
- h. One of the thermal shield circulation pumps is running and normal flow is established in the thermal shield loop.
- i. The temperature of the primary D₂O is in the range of 75 to 85°F, depending upon the cooling tower water temperature. The normal circulation rate has been established in the cooling tower circuit.
- j. All appropriate shield plugs, irradiation experiment casks, and shields around beam experiments are in place.
- k. All instruments and monitoring stations show normal levels of radioactivity, temperature, pressure, and flow in all of the process systems.
- l. The building exhaust system is operating normally, and the building containment shell is tight.

The neutron sensing instruments will all show "on-scale" readings due to the γ -n reaction in D₂O from the irradiated fuel element fission products. The excess reactivity which the core would have at this point if all rods were fully withdrawn is 8.5% k.

To start the reactor the operator first withdraws the main rod crown under manual control. (The eight main rods will be occasionally referred to as a "crown", as will the set of auxiliary rods.) The operator withdraws the main rod crown to the expected critical position. The reactor will still be subcritical at this point because the auxiliary rods will have remained fully inserted. Next, the auxiliary rod crown is withdrawn in steps by the operator until indications of a positive period show on the appropriate meters and on the safety channel trip meters.

The operator controls the auxiliary rod crown during the power rise to trim the rate of rise to a suitable value. As the reactor power approaches the desired operating level, the operator trims the period to infinity with the auxiliary rod crown and then switches to automatic power level control.

The automatic control, in turn, operates a single auxiliary rod. With the reactor on automatic control, the flux sensing system sends signals to the auxiliary rod drive under automatic control. This drive moves the rod in such a fashion as to maintain a constant flux in the reactor midplane. As xenon grows in, fuel burnup occurs, and as experiments are put into and taken out of the reactor, the "automatic" rod will cycle in and out. The net motion over a reasonably long period of time will be out, to compensate for burnup of fuel. When the automatic rod has made a net outward motion of 1 in. from the nominal position of the auxiliary rod crown, a signal is given to the operator. The operator then manually withdraws one of the other fifteen rods 1 in. This withdrawal will be in short "jogs" and the auxiliary rod under automatic control will balance the reactor power and maintain constant flux during this manual operation. The automatic rod is then returned to its initial position and will start cycling outward again. Repetition of this pattern of control will result in the two rod crowns being slowly withdrawn, always in symmetric positions about the reactor midplane, toward the full-out positions at the end of the operating cycle.

5.2.2 Shutdown Procedure. The normal shutdown proceeds as follows. The main and auxiliary control rods are driven to the fully inserted positions at normal speed. All of the process systems are maintained at full flow until all

temperatures have stabilized at the new equilibrium values appropriate to the much reduced power being generated in the core. Thereafter the main D₂O pumps are shut down and the cooling tower water flow reduced accordingly. The core cooling flow is maintained by the shutdown pump. Flow is reduced as appropriate in the thermal shield system and in the experimental D₂O system. The primary system pressure is maintained for 12 to 16 hours, during which time the fission product afterheat decays to about 750 KW. In preparation for entry into the reactor pit and fuel changing operations, the helium blanket gas is vented and reduced to atmospheric pressure.

5.2.3 Startup with Clean Core. The outline of the normal startup and power operation given in Section 5.2.1 assumes a core loading of 14 perimeter elements which are partially burned up and 14 inner elements which are unburned. For this normal configuration core the peak-to-average power density ratio is 2.82, and the maximum power density and heat flux at 40 MW are 1.31 KW/cm³ and 1.09 x 10⁶ BTU/hr-ft², respectively.

In a clean core of all unburned fuel elements, the peak-to-average power ratio is increased to 3.5 (see Section 3.3.2). Thus, if the normal maximum power density and heat flux are not to be exceeded, the clean core power level must be reduced in the ratio

$$\frac{2.82}{3.5} = 0.806,$$

or to approximately 32 MW. After 7 to 9 days of operation at 32 MW, the perimeter elements will have reached a burnup corresponding to that normally present at startup, and the power level can be raised to 40 MW for the remainder of the half-cycle.

A second important feature of the clean core startup is the reduced neutron level in the shutdown configuration. The photoneutrons from fission product gamma radiation are not present in the clean core case. There will, of course, be photoneutrons produced by the decay gammas from the steel thermal shield after the reactor has operated at 40 MW for a half-cycle, but even this source is absent in the initial

startup. It is necessary, then, for clean core startups to provide a supplementary neutron source to insure adequate counting rates in the two fission chamber startup instrument channels. The supplementary source will be placed in one of the in-core irradiation thimbles.

The clean core startup and power operation will follow the sequence of Section 5.2.1 except that a supplementary neutron source will be used, and the power will be held to 32 MW for the first part of the half-cycle. The supplementary source will be withdrawn from the core region during the startup, when the reactor becomes critical.

It should be noted that even for 40 MW operation with a clean core, the maximum heat flux is only 1.35×10^6 BTU/hr-ft², compared to the steady state burnout heat flux of 4.76×10^6 BTU/hr-ft². Operation of the clean core at the full 40 MW power does not, therefore, constitute a hazard to the reactor system.

5.2.4 Fuel Changing Operations. At the end of each operating period, the reactor is shut down and the outer 14 fuel elements in the core are discharged. The remaining 14 elements are moved to the outer positions, and new elements are charged into the center of the core.

All fuel additions, removals, and rotations are approved in writing by either the Reactor Division Head, or the Operations Group Leader, or a designated Supervisor. This information is provided to the shift crews by means of a special work sheet which carries the authorized signature. The responsible staff member signing the fuel loading form provides assurance that changes to the loading have been considered from the viewpoint of reactivity effects and coolant requirements.

The core is cooled for 12 to 20 hours before any fuel elements are discharged to reduce the fission product decay heat. At the end of this cooling period, the temperature of the primary coolant is reduced to approximately ambient air temperature, the reactor tank is depressurized, and the fuel handling equipment readied for use. The D₂O level in the reactor vessel is raised to increase the water height over the core for shielding. A top shield plug is removed to permit entry to the reactor pit. The top plate of the vessel is unbolted and removed.

The discharging procedure is as follows. The tritium container is set in place over the vessel and discharge chute (see Figures 39 and 40). With the shielding shutters open, a fuel element to be discharged is lifted out of the core by the grapple. The indexing mechanism is moved until the lifting cable is lined up with the groove in the shutters, and the shutters are closed. The element is lifted out of the D_2O and clear of the top of the tank. The cooling air bonnet is arranged so that the element aligns itself and slips up into the bonnet as it leaves the D_2O . When the element is clear of the water the cooling air is turned on. The slide is moved into alignment with the top of the discharge chute, and the element is lowered into the chute and released to the canal.

After the 14 outer elements are thus removed from the core, the 14 inner elements are moved to the new core locations and 14 fresh elements are charged by hand from the reactor pit.

5.2.5 Beam Experiment Operations. The installation of beam experiments will be made during reactor shutdowns. Most beam experiments will not require access to the portion of the tube that lies on the core side of the shutter. The shutter can be closed, therefore, affording ample shielding. The aluminum seal cap which is part of the outer stepped plug need not be removed, so that containment of the CO_2 atmosphere between the vessel and thermal shield remains intact.

Should special radiators or scattering samples be required in the beam tube closer to the core, a shield plug will have to be provided for installation with the shutter open. The plug will be fitted with a seal cap and will have D_2O sample cooling lines cast into it.

During operation of the reactor limited access to collimators in the outer stepped plug, or adjustment of apparatus is possible with the shutter closed. Discharging a collimator requires a special cask for transport to the beam plug storage vault on the experimental floor.

5.2.6 Irradiation Thimble Operations. These vertical facilities can accommodate capsules, either with or without electrical and sampling leads. In either case, the sample capsule with appropriate attachments is mounted into a handling cask. The handling cask is then bolted onto the bedplate at which an irradiation thimble terminates. The cask is shown in Figure 38. With the cask forming a sealed, pressure-tight extension of the thimble itself, a ball valve in the thimble line is opened to permit the sample to pass down the tube and into position for irradiation. The cask may be maintained dry during the irradiation period by pressurizing to a few pounds gauge with helium, or the experimental D_2O coolant may be allowed to rise into the cask as is appropriate in individual cases.

When the irradiation is completed the sample is withdrawn into the cask with its attached wires being taken up on an internal reel. The ball valve in the thimble line is closed, and D_2O above the ball valve is drained. Samples without leads may be released from the holding collet and rotated into a position from which they may be withdrawn into a secondary cask. Samples with leads attached will remain in the handling cask. The cask will be unbolted and transported to a hot cell where the sample may be removed and prepared for further experiment.

5.2.7 Fuel Storage. Fresh fuel elements are stored prior to use in a storage vault on the Operations Floor. The racks will have a capacity for 200 to 300 elements in "critical-safe" racks around the walls of the vault.

Used fuel elements discharged from the reactor will be stored in the canal in the basement of the building. After discharge from the reactor, the end box aluminum sections of the elements are removed and the fuel sections are placed in storage racks. The racks are constructed of boron stainless steel to insure a subcritical condition of the used fuel in any array. Space is available in the storage canal for 300 elements, or more than one year's total supply for the reactor.

5.3 Coordination of Experiments

The procedures to govern the placing of experiments in the reactor will be based on the experience which has been gained in operating other Brookhaven research reactors. A sequence of experiments with the critical mock-up of the HFBR will establish the reactivity effects of a wide range of possible experimental components (irradiation samples, cooling tubes, collimators, etc.). These will show the point beyond which a complete hazard analysis will be needed before the experimental change is made. As operation of the reactor goes on, the knowledge of the reactivity effects of experiments will, of course, improve.

It is expected that no hazard approval outside the Reactor Division will be required for small sample irradiations.

The precise procedures which will apply will be included in the Operating Manual and the final Hazards Summary Report, both of which will be issued for approval before startup.

5.4 Maintenance

5.4.1 Control Rods and Drives. The control rod mechanisms are divided into two basic sections. The first is the drive mechanism itself which is exterior to the reactor vessel. The second section is the support tube structure within the vessel which terminates in the poison blades. The drive mechanisms may be reached upon reactor shutdown for maintenance. For major repairs or overhauling the drive mechanisms may be disconnected and spares substituted.

Maintenance on the portions of the unit inside the vessel will require that the support tube structure be unbolted from the reactor vessel and that the entire structure be withdrawn upward through the fuel handling opening in the top of the vessel. In this withdrawal the main rod and associated auxiliary rod below it will come out with the support tube as a single unit. Since the lower ends of these pieces will be intensely radioactive, the support tube structure must either be handled remotely and passed through the fuel discharge chute to the canal, or must be brought out into a special shielding cask.

5.4.2 Reactor Vessel Internal Structure. The internal structures include the rod drives, the upper and lower shrouds, the grid plate, and the irradiation thimbles. The drives have been discussed above. All of the internal structures of the vessel may be removed after the fuel has been discharged to the canal. The irradiation thimbles may be pulled up through the holes in the vessel top after the upper sections have been disconnected. The vessel top works may be removed to provide a clear opening of 46 in. I.D. in the neck of the vessel. The shroud may be unbolted from its flange and withdrawn upward out of the vessel. The lattice plate may then be unbolted from above, tipped on edge, and removed from the vessel. All of these items will, of course, be radioactive after the reactor has run for some time, and extensive shielding will be required in their removal.

5.4.3 Primary System Components. Aside from the reactor vessel itself, the primary system components are contained in shielded cells in the building basement. These cells are arranged so that one of the two parallel loops may be valved out of the system. With one loop operative, the reactor may be run at from one-half to two-thirds full power while maintenance work proceeds on the pump and heat exchanger in the "down" loop.

Maintenance work on the main flowmeter or the shutdown heat exchanger, EA-103, will require that the reactor be shut down and the core discharged to the canal. Since the shutdown pumps are in parallel, one of them may operate while the other one is being maintained.

5.4.4 Fuel Handling Apparatus. With the exception of the tritium container hood, which has no working mechanism except the simple fuel element slide, all of the fuel handling apparatus is normally outside the shield during reactor operation. It is, therefore, readily available for all necessary maintenance.

5.4.5 Experimental D₂O System. Individual experimental facilities may be removed from the reactor during a shutdown, the piping to them subsequently closed off with valves, and the remaining facilities will continue to function during the succeeding period of reactor operation. In general, the installation or removal of an irradiation thimble can take place only when the reactor is shut down. The same applies to the installation of scattering targets in the beam holes inboard of the seal caps.

In the experimental D₂O process system, there are two full capacity pumps, which permits one to drive the system while the other is being serviced. Since there is only one heat exchanger, maintenance on it requires that the experimental system be down.

5.4.6 D₂O Cleanup System. The main D₂O cleanup beds and filters are installed in deep pits in the basement floor. Individual units which are to be removed and replaced may be valved out of the system, disconnected, drained of D₂O, and removed from the building in shielded containers. An overhead monorail system of 15 tons capacity is provided for the handling of these beds and filters.

6. RADIOACTIVE WASTES AND EFFLUENTS

6.1 Gaseous Wastes

In normal operation the only gaseous radioactive wastes of concern are tritium and argon-41. The tritium, which is held in water molecules of the type TDO, appears in the exhaust gas system from fuel handling operations and irradiation sample handling which evaporate D₂O. The primary D₂O will build to an equilibrium tritium content of about .018 C/cm³ D₂O in about 40 years (35). We are concerned both with the total annual addition of tritium to the atmosphere, and with the maximum expected concentration at any one time.

The fuel handling operations are expected to evaporate about 2 pounds of D₂O per cycle, all in the space of four hours. This means that at equilibrium concentration a maximum amount of 3.71 curies of tritium will be expelled from the stack per hour during the four hour fuel change. The normal stack exhaust is 4.82×10^5 m³/hr, so that the tritium concentration at the stack top in these peak concentration periods is 7.7×10^{-6} C/m³. Since this concentration is well below the 20×10^{-6} C/m³ 40 hour week tolerance level (36) even at the top of the stack, ground level concentrations will be negligible.

The total tritium discharge per year due to both fuel handling and sample handling is expected to be that contained in 72 pounds of D₂O, or 535 curies at equilibrium tritium content.

Argon-41 appears in the reactor pit by activation of the naturally occurring argon-40 in the upper pit air atmosphere. (The lower pit has a CO₂ atmosphere.) The activating thermal flux in the upper pit, due entirely to fast neutron leakage from the core region, is about 1.8×10^5 cm⁻²-sec⁻¹. The macroscopic argon activation cross section is 1.34×10^{-7} cm⁻¹, yielding 2.4×10^{-2} argon-41 nuclei per cm³-sec. It is convenient to consider a zero purge rate in the pit, since this yields the maximum concentration. This maximum is 6.5×10^{-7} C/m³ at equilibrium which is about 1/3 the 40 hour week occupational tolerance for an infinite atmosphere (36). To obtain an upper limit on the discharge concentration we assume that the equilibrium concentration is flushed from

the pit in one hour. The upper pit has a net volume of about 70 m^3 . The normal stack exhaust flow of $4.82 \times 10^5 \text{ m}^3/\text{hr}$ yields a stack top concentration of $.95 \times 10^{-10} \text{ C/m}^3$, about 4000 times less than tolerance even at that point. Ground level concentrations are further reduced by a factor of several thousand in the worst case.

The total equilibrium activity of argon-41 produced in the upper pit is $46 \mu\text{C}$, so that this is the maximum load which could be imparted to the atmosphere during operation of the reactor.

The gaseous radioactive effluents expected from the HFBR, as calculated above, may be compared to those of the existing reactors at Brookhaven. The Graphite Reactor releases about 16,000 curies/day of argon-41, with a resulting equilibrium load of 1,770 curies. The release is from the same stack which the HFBR will use. The Medical Reactor releases about 3.5 curies of argon-41 per megawatt-hour of operation from a 150 ft high stack. Since the operation is intermittent, to suit experimental needs, there is no regular release rate. The average site boundary dose rate for 1960, at the monitoring station showing the highest activity, was 0.72 mrad/week , or about 7% of tolerance.

6.2 Liquid Wastes

Liquid wastes may be divided into two classes: those which contain traces of reactor D_2O , hence tritium, and all other liquid radioactive wastes. The non-tritium bearing wastes are collected throughout the reactor building in standard BNL "D" and "F" waste drain systems connected to waste storage tanks in the reactor building basement. Periodic transfer of liquid wastes from these vessels is made through an alloy steel pipeline to the BNL Waste Treatment Area. There it is stored with other Laboratory radioactive liquid wastes, eventually distilled, and the discharged concentrate used as a slurry in making concrete for the vaults in which solid radioactive waste is packaged for final disposal. D_2O waste, which contains tritium, will be collected separately. A final disposal method for the tritium-bearing waste water has not been decided upon, but it is likely that this water will also be used to make concrete for the waste storage vaults.

6.3 Solid Wastes

The major solid radioactive waste from the reactor will be the spent fuel elements. Fuel elements will be removed from the reactor every three weeks and transferred to the storage canal. It is expected that the spent fuel elements will be held in the canal for periods of at least six months to permit decay of all but the long-lived fission products. Periodic shipment of spent and decayed fuel elements in shielded casks will be made to the Idaho Falls Chemical Processing Plant.

Activated trash with an activity of less than 50 mR per hour is collected in radioactive trash cans located about the building. Solid waste of higher level from the irradiation experiments and from reactor operations will be transferred in shielded containers to the waste storage area. The normal BNL procedure for disposal of these wastes is to store them in concrete shielded vaults.

6.4 Fission Product Inventory

Table XXIV, below, summarizes the fission product inventory in the average fuel element, the peak power element, and the whole core 100 seconds after shutdown from the 40 MW normal power level (37).

Table XXIV

HFBR 40 MW Fission Product Inventory
100 Seconds After Shutdown (40 day operation)

	<u>Average Element</u>	<u>Peak Element</u>	<u>Total Core</u>
Iodines	3.40×10^5 C	4.80×10^5 C	9.54×10^6 C
Bromine	5.10×10^4	7.20×10^4	1.52×10^5
Total halogens	3.91×10^5	5.52×10^5	9.69×10^6
Xenon	2.39×10^5	3.40×10^5	5.40×10^6
Krypton	1.45×10^5	2.07×10^5	4.07×10^6
Total rare gases	3.84×10^5	5.47×10^5	9.47×10^6
Solid f.p.'s	2.43×10^6	3.44×10^6	7.05×10^7
Total f.p.'s	3.20×10^6	4.54×10^6	8.97×10^7

7. SAFETY ANALYSIS

7.1 Introduction

The safety analysis of the HFBR has been carried out along the following lines. First, the kinetic behavior of the reactor has been examined in detail. A summary of the results is presented in Section 7.2 to establish the response of the reactor to various kinds of reactivity inputs. In this analysis it will be shown that, in accordance with calculations made since the time the Site Report was written, the HFBR negative void coefficient of reactivity is very effective in limiting the magnitude of transients. With the transient response of the reactor established, the various kinds of reactivity accidents which may happen are considered in Section 7.3. Equipment failures and malfunctions are considered in Section 7.4. The environmental hazards are reviewed in Section 7.5 with regard to the building, the reactor structures, and the location.

In Section 7.6 the maximum credible accident is examined and its consequences in terms of radioactive dosage to the general public are calculated.

Finally, in keeping with the recent Reactor Site Criteria issued by the AEC, accidents beyond those normally considered credible are considered. The exclusion area, low population zone, and population center distance for these latter accident cases are calculated and are presented along the lines suggested in the Reactor Site Criteria.

7.2 Transient Behavior

7.2.1 Theoretical Model

A. Origin of the Model

Since the time of the Site Report, the transient behavior of the HFBR has been examined in detail. There is now a better understanding of the response of the reactor to very severe reactivity inputs.

The theoretical analysis is based on a model which has been shown to predict transient behavior in good agreement with that observed at SPERT III.

SPERT III is a reactor at the NRTS, built and operated by the Phillips Petroleum Corporation. It is used to provide data on fast transients in water cooled and moderated reactors of the ETR type. The fuel elements are flat plates of clad U-SS alloy, gathered into a box structure. Light water serves as both moderator and coolant, being circulated through the core at the moderately high rate of 18 ft/sec. The flow direction is up, and after passage through the core the coolant flows downward through a thin reflector region between the tank wall and the core boundary. Although there is coolant capacity for high power operation, the power levels prior to transients have so far been kept low.

The SPERT group was kind enough to supply detailed data on three transients, characterized by reactivity step sizes different enough to lead to quite varied behavior. The lowest reactivity led to a moderate power increase without boiling. The next caused much greater power generation, but still no boiling. The third was severe; the reactor power rose to a peak value of nearly a thousand megawatts, and considerable steam generation occurred. The experimental data supplied were principally curves of power and fuel temperature during the transients. In addition, there were curves of the steady state power distribution in the core (the control rods were partly inserted), the neutron lifetime, and the uniform void coefficient of reactivity.

The detailed spacial dependence of the void coefficient and of the water and fuel temperature coefficients were calculated at Brookhaven, the input data being details of the core and control structure, the uniform void coefficient, and the experimental power distribution.

Unfortunately, the power distribution is not sufficient for a proper calculation of the spacial dependence of the reactivity coefficients. The dominant moderator effects on reactivity are due to neutron scattering and moderation, so that one actually requires detailed knowledge of the neutron flux and adjoint as functions both of energy and position in the core. This information would be difficult to get. Estimates have, therefore, been based on simple flux-squared weighting.

The errors so introduced are of higher order, because the uniform coefficients are still correct. The effects of the faulty assumptions are only felt through the non-uniform distributions of temperature and steam during the transients, and through the role of the weighting function in establishing an effective core size for the reactor.

The fuel coefficient was calculated from the effect on water displacement. Part of the coefficient is weighted, because it is associated with the displacement of water from channels as the fuel expands. Part is not weighted, because it reflects an over-all change in core size as the temperature rises. In this connection, it was noted that the core expansion, in effect, moves fuel farther up into the region of partial control rod insertion. The effect on reactivity was calculated from the control rod calibration curves; the estimated effect is the larger part of the metal coefficient.

The reactivity coefficients which were calculated as just described were used as feedback parameters in an augmented version of the conduction-boiling model of transient behavior.

The conduction-boiling model was developed by the SPERT group to account for some of the features of the transient performance of SPERT I. SPERT I resembles SPERT III in the nature of the fuel and the moderator. It is not force-cooled and the core rests in a quiescent pool of water. Heat transfer to the water during a transient takes place by normal conduction, since there is not enough time for convection currents to be set up.

The augmented version of the conduction-boiling model which has been developed at Brookhaven to account for the transient behavior of SPERT III and of the HFBR differs in the nature of the heat transfer from the fuel to the water. The mechanism is assumed to be film conduction. Also, the augmented model has been set up as a time-dependent set of equations, so that the detailed course of the transient can be calculated. The equations, then, take into account the transport of coolant water and steam through the core during the excursion.

As in the early version on the conduction-boiling model, the steam is created after the fuel plate surface temperature reaches the boiling point of the water, and the steam generation is assumed to be proportional to the amount of heat being transferred to the water while the fuel temperature remains above this point. The fraction of heat causing steam formation is called the steam generation factor, f_{as} . In SPERT I analyses, f_{as} was found to fall between 10^{-2} and 10^{-3} .

In both the original and the augmented model, steam void collapse is neglected. In the augmented model, the steam generated is assumed to be transported along with the water in the coolant channels, until it vanishes at the core outlet.

Point reactor kinetic equations are used.

B. The Equations

Heat Equations

$$\rho(x,y,z,t) = C_F \frac{\partial}{\partial t} T_F(x,y,z,t) + hA [T_F(x,y,z,t) - T_W(x,y,z,t)]$$

$$C_W \frac{\partial}{\partial t} T_W(x,y,z,t) = hA [T_F(x,y,z,t) - T_W(x,y,z,t)] - VC_W \frac{\partial T_W}{\partial z}(x,y,z,t)$$

$$C_F \frac{dT_R(t)}{dt} = VC_W \left[\int d x \, d y \, T_W(x,y,z_0,t) - A_C T_R(t) \right]$$

Steam Void

$$\frac{\partial}{\partial t} m(x,y,z,t) = -V \frac{\partial}{\partial t} m(x,y,z,t) + \mu(x,y,z,t)$$

$$\mu = 0 \text{ if } T_F < T_B$$

$$\mu = \frac{f_{as}}{L} \rho(x,y,z,t) \text{ if } T_F \geq T_B$$

Compensated Reactivity

$$\rho_C = \rho_W + \rho_B + \rho_F + \rho_R$$

$$\rho_W = \int dV w(x,y,z) \alpha_W [T_W(x,y,z,t) - T_W(x,y,z,0)]$$

$$\rho_B = \int dV w(x,y,z) \alpha_B m(x,y,z,t)$$

$$\rho_F = \int dV \alpha_F [gw(x,y,z) + (1-g) \frac{1}{V_C}] [T_F(x,y,z,t) - T_F(x,y,z,0)]$$

$$\int dV w(x,y,z) = \frac{1}{V_C}$$

$$\rho_R = \alpha_R [T_R(t) - T_R(0)]$$

Kinetic Equations

$$\frac{dP}{dt} = \frac{1}{\ell} [(\rho_0 - \rho_C - \beta)P + \sum \lambda_i F_i]$$

$$\frac{dF_i}{dt} = \beta_i (1 + \rho_0 - \rho_C) P - \lambda_i F_i$$

Meanings of Symbols:

p = power density

C_F = fuel heat capacity per cm³ of core

T_F = fuel temperature

T_W = water temperature

h = heat transfer coefficient

A = heat transfer area per cm³ of core

C_W = water heat capacity per cm³ of core

C_R = heat capacity of reflector

T_R = temperature of reflector

A_C = core outlet area

V = water speed in channels

m = steam mass per cm³ of core

f_{as} = steam generation factor

L = latent heat of vaporization
 ρ_C = compensated reactivity
 ρ_W = reactivity compensated by water temperature change
 α_W = uniform core water temperature coefficient
 ρ_B = reactivity compensated by steam
 α_B = uniform void coefficient
 ρ_F = reactivity compensated by fuel temperature change
 α_F = uniform fuel temperature coefficient
 w = weight factor
 g = fraction of metal coefficient which is weighted
 V_C = volume of reactor core
 ρ_R = reactivity compensated by reflector temperature change
 α_R = reflector temperature coefficient
 P = reactor power
 l = prompt neutron lifetime
 ρ_0 = applied reactivity
 β = delayed neutron fraction
 β_i, λ_i = delayed neutron parameters
 F_i = delayed neutron precursor content in energy units
 Water flow is in the $+z$ direction.
 The core outlet is at $z = z_0$.

C. Methods of Calculation

The above equations have been coded both for analog and digital computers. Both kinds of calculational methods have their advantages; the analog computer yields results sooner, and it treats the time behavior in a continuous way rather than approximating it by difference equation methods. The spacial aspects of the problem are best taken care of by the digital method, which allows a multiplicity of spacial regions to be used. The resulting advantage shows up in the detailed analysis of the way the temperatures change and the steam grows in at different points when temperature peaking is strong.

The analog code assumed the reactor core to be divided into four horizontal slices. Average temperatures, power densities, and steam void contents were calculated in the different regions. Water and steam transport were accomplished by passage of the average quantities in one region to the next.

The digital code calculations were done with an LGP-30 computer. The reactor core was divided by a rectilinear grid into twenty sets of horizontal regions, each set consisting of as many as ten possible radial regions. Thus, there were as many as 120 core regions. The code calculated the inlet and outlet values of water and fuel temperature and of steam density. Transport occurred through transfer of outlet parameters. One vertical set of regions was devoted to the hot channel, which also included the hot spot.

D. Application to SPERT III Transients.

The three transients whose histories were calculated were initiated by step reactivity inputs inserted at low power. Their characteristics were:

Transient No.	I	II	III
ρ_0	0.0082	0.0087	0.0102
Uncompensated period (ms)	39.69	23.27	10.08

Two values of f_{as} were used in connection with the calculation of transient III (the only one which boiled water); it was found that a full parametric survey would not have been useful, because there is no obvious criterion by which a best value could be selected. The reactivity coefficients used were the ones which had been calculated from the reactor properties; no adjustments were made. Similar remarks apply to all other parameters, such as the heat transfer coefficient.

Calculations were done both by analog and digital methods with $f_{as} = 0.002$ and 0.005 . Figures 53, 54, 55, and 56 show the results of the analog calculations of power vs. time, and Figures 57, 58, 59, and 60 show the results of the digital calculations. In each case, the experimental curve is also shown, so that comparison may readily be made.

Two conclusions may be drawn from the results. The first is that the two kinds of calculations give results which are in acceptable agreement. The differences which exist are the results of the coarser spacial mesh of the analog treatment.

The second conclusion is that the transients in which no steam is generated are reproduced quite well by the calculations, and that the choice of $f_{as} = 0.002$ leads to good agreement with experiments when boiling occurs.

The agreement is best in the rising part of the power curve. After the power peak, while temperatures are still rising and the reactivity compensation is still increasing, the calculated curves show a tendency to lag behind the measured ones. Beyond the power minimum where the coolant flow has removed reactivity compensation this tendency continues. The differences here are most likely the result of the way the weighting function is chosen. A selection which made the apparent height of the reactor smaller would lead to more rapid adjustment of reactivity in these time regions, where transport effects establish the time scale of the water temperature and steam readjustment.

A choice of $f_{as} = 0.005$ would lead to somewhat better agreement between the falling region of the measured and calculated power curves. The calculated power dip would, however, be much too low, indicating the generation of too much steam. It is concluded that $f_{as} = 0.002$ is a more appropriate selection; it leads to better agreement in the mean, and it is also more conservative in that it leads to higher power generation rates and higher fuel and water temperatures.

The experimental data also include the temperature history at the hot spot. These measurements are not so well reproduced by the calculations, as can be seen from Figures 61, 62, and 63 (which show the comparison between digital results and measurements). No doubt adjustment of the heat transfer coefficient and to some extent the reactivity coefficients would improve the agreement. It has been believed preferable, however, to use only the values of these parameters which are believed to be correct. In any case, the calculated fuel temperatures are greater than the measured ones, and presumably this point introduces conservatism into the application of the model to accident analysis.

E. Conclusions Regarding the Model.

The foregoing has shown that the theoretical model predicts with reasonably good accuracy the time behavior of the power of a light water moderated reactor which is cooled by forced convection. The fuel temperatures are overestimated by the calculations. The analog and digital codes lead to comparable results.

SPERT III and HFBR have both similarities and differences. The geometry of fuel and moderator is the same in the two, and so identical treatment of the heating and cooling is appropriate. The one difference here is in the shape of the power distribution. The power peaks strongly at the boundary of the HFBR core, whereas the maximum power generation in SPERT III takes place at the core center.

The same kinetic equations are likewise applicable, at least within the point reactor approximation. The core reactivity coefficients are nearly the same. HFBR has a very large reflector coefficient, which is not important for fast transients, but which acts as a damping force for longer time transients. The prompt neutron lifetime in the HFBR is about thirty times that in SPERT III. This is, in fact, the most striking difference between the two systems, because it will lead to much slower response to reactivity changes in the heavy water moderated system.

Because the fuel coolant arrangements are so nearly identical, similar values of f_{as} will apply to steam production.

Altogether, it is to be expected that the theoretical model, which yields results in good agreement with observations at SPERT III, should be applicable also to the HFBR. The discussion of transients which follows is based on this premise.

7.2.3 HFBR Transients: Introduction. An extensive set of calculations of HFBR transients has been performed, with both step and ramp reactivity inputs. Digital methods were used for some, and analog methods for others, according to which seemed most appropriate. In many cases, both methods were used, so that the agreement might be checked. The constants used in the calculation, including reactivity coefficients, are given in Table VII.

Transients were calculated both from high and low power initial states, because the behavior in the two cases is different. Only the action of inherent safety properties was included in the calculations. That is, reactivity compensation through the appropriate feedback mechanisms was assumed to exist, but no action of control rods was taken into account.

The power distribution used for the calculation was that measured in the critical experiments. The weighting function for the water and steam coefficients, and for a fraction of the metal coefficient, was calculated by perturbation methods with the GNU Code which has been used for the design studies.

An initial survey gave the short-term response to step reactivity input induced transients, as a function of the step size. The reactor power generally goes through a peak after the introduction of a reactivity step, and then falls again as steam grows in. The short-term response was calculated up to and a little past the power and temperature peaks to show what maximum values are assumed by these variables. The time intervals covered are usually under one second, and are, in fact, so short that the negative reactivity effect associated with the rise in reflector temperature has not had enough time to make itself felt very strongly.

Three calculations were done for different step sizes to show how the reactor would behave over a period of many seconds. The reflector reactivity effect is the dominant feature of these transients.

Finally, calculations were made to establish the behavior of the reactor as reactivity is added at a steady rate.

7.2.3 HFBR Transients: Short-term Step Response. As stated above, the calculations assume that a step reactivity input to the reactor occurs. The power rises, and as it does so the fuel and water temperature increase. When the fuel plate temperature reaches the boiling point (190°C), steam generation begins to take place. As the water is heated, and steam is formed, water and steam are transported at a steady rate through the core and out into the reflector. This heats the reflector. Eventually the temperatures and the steam content reach such a value that the associated reactivity compensation stops any further power rise. Roughly speaking,

for larger transients this point of maximum power occurs a short time after the compensated reactivity is equal to the amount by which the initial reactivity exceeds the delayed neutron fraction. For smaller reactivities, the point of peak power is not so readily defined.

Each transient whose history is discussed in this section is characterized by a power peak of this sort. The calculations extend beyond the power peak, at least to the point at which the maximum fuel temperature occurs.

Figures 64, 65, 66, and 67 show the results of a selected set of digital calculations, with reactor power initially at 10 KW and with reactivity step sizes ranging from 1.0 to 1.56 % in k. The quantities shown in the figures are the total reactor power, the compensated reactivity, and the fuel temperature at the hot spot (the hot spot does not shift position during the transient).

Figures 68 and 69 give the maximum values achieved by the reactor power and the hot spot fuel temperature at their respective peak values, as a function of the size of the reactivity step. Curves are given for two values of the steam generation factor f_{as} , 0.002 and 0.005. As expected, the value 0.002 leads to greater peak powers and fuel temperatures. Since $f_{as} = 0.002$ seems to fit the SPERT III transient, and since it predicts the larger transients, the safety analysis is based on this choice.

Figures 70, 71, 72, and 73 show selected transients similar to the preceding, but differing in that the reactivity step is assumed to take place when the reactor power is initially at the design point of 40 MW. Figures 74 and 75 give the peak values of power and hot spot fuel temperature, again neglecting steam blanketing, as functions of the reactivity step size.

It is seen that transients initiated at high power reach higher peak powers than those which begin at low power. In either case, a striking aspect of the transients in the HFBR is their consistently smaller size for a given reactivity input than corresponding transients of a light water reactor such as SPERT III. The reason for the difference is not the size of negative reactivity coefficients; these are quite

similar in the two reactors. Rather, it is the long prompt neutron lifetime of the HFBR which is the cause. The history of the "first pulse" response is accordingly much longer in the HFBR. As an illustrative example, one may compare the transient of Figure 59 (SPERT III) and Figure 64 (HFBR), both being transients initiated at low power by approximately the same reactivity inputs (1.0 % in k). The time from 1 MW to peak power is about 80 ms for the SPERT transient; it is about 1850 ms for the HFBR transient.

The more sluggish response of the HFBR causes the reactivity feedback effects to lag behind the instantaneous power level by a relatively shorter time. Thus, the peak power and the peak compensated reactivity are not very far apart in time in the HFBR, while they are well out of phase in a SPERT transient.

The two features--the difference in rapidity of response and the difference in peak power generation--tend to cancel, so that the two kinds of reactors can tolerate step reactivity inputs of comparable amounts.

7.2.4 HFBR Transients: Long-time Step Response. It is important to establish that the power and temperature history after the initial pulse is not more dangerous than that at the peak of the pulse. Analog computer runs have, therefore, been made to establish the behavior of the reactor power, the fuel temperature at the hot spot, and the compensated reactivity over times long compared with the initial reactivity burst.

The results are given in Figures 76, 77, and 78 for step reactivity inputs of approximately 0.4%, 0.8%, and 1.2% k (0.50 \$, 1.00 \$, 1.50 \$). Calculations were extended in each case a little beyond 20 seconds. As in all other accident analysis described above, none of the safety rods were assumed to operate.

The curves show an interesting result. After the initial pulse response of the reactor, which does not recur, the power and the fuel temperature settle down quickly to new constant values which are greater than those initially present but less than those experienced at the peak of the transient.

It follows that the worst conditions are met at the peak of the initial pulse. A step input which does not melt fuel in this pulse will not melt fuel afterwards.

The result found here is entirely due to the large reflector temperature coefficient. Before a second pulse of power can begin, the reflector temperature begins to rise at a rate fast enough to damp it out. This can be seen from the curves of reflector temperature given in Figures 76, 77, and 78.

The long-term transient response of the HFBR must, therefore, be viewed as falling somewhere between that of normal solid fueled, water cooled reactors and homogeneous reactors. The large uniform temperature coefficient of the HFBR places it in this range. The reflector coefficient, which is the largest part of the uniform coefficient, is delayed by several seconds. This delay permits the occurrence of a first pulse of power after the reactivity step, but the delay time is short enough to prevent further pulsing.

It must be pointed out that the fuel temperatures shown in Figures 76, 77, and 78 are not the temperatures at the hot spot. The hot spot occurs at the corner of the core exit surface. The analog computation yields essentially the average fuel temperature of all fuel within 0.5 in. of the core exit. The relation between the hot spot temperature T_m , and the peak temperature T_a calculated by the analog program is very nearly constant during the transient; it is

$$(T_m - 50) = (1.72)(T_a - 50)$$

the value 50°C being the inlet water temperature and, therefore, the reference temperature for all calculations.

One last point is worthy of mention here. The 1.50 \$ transient shows evidence of oscillations at two points following the initial pulse. These oscillations are the result of growth and collapse of steam generation in the large center region of the reactor. The computer treats this large volume as only two discrete regions, and the steam generation, therefore, takes on an artificial on-off character conducive to relaxation oscillations. The same oscillations have occurred at the same power levels in analog calculations of

other ramp induced transients. A digital computer recalculation of one of these transients showed small power oscillations (about 0.1% of the power in magnitude) at nearly the same power levels indicated by the analog computation. It is concluded that a minor region of instability exists in the region above about 150 MW of power, that the instability is caused by the shape of the power distribution, and that the analog code with its coarse spacial mesh overestimates the size of the instability. The region discussed is so far above the operating region that no further study of the instability has been made.

In summary, the calculations reported in this section establish the point that the initial power pulse following a step reactivity input contains the interval of peak power and peak fuel temperature. The slow reflector temperature change damps out successive pulsing.

7.2.5 HFBR Transients: Ramp Response. The reflector temperature and reactivity feedback are also of prime importance in determining the nature of the response of the reactor to a steady, ramp reactivity addition. Figure 79 shows the behavior of the reactor power, the hot spot fuel temperature, and the reflector temperature as reactivity is added at the steady rate of 0.10 \$/sec (0.078 % k/sec) to the reactor from an initial power level of 1 MW. The calculation was done with the digital computer. Reactivity was added steadily for 50 sec, for a total of 5.00 \$, or 3.9 % in k.

The detailed calculations show that the maximum uncompensated reactivity occurs at about $t = 10$ sec, where 1.00 \$ total has been inserted. From this point on the rising reflector temperature causes the uncompensated reactivity to decrease steadily.

The power rise shows no tendency toward an initial burst. After increasing sharply at about the prompt critical reactivity, the power tapers off into a slow, linear rise with a doubling time of about 27 sec.

The reactor response to other ramp rates started at low initial power is much the same as that shown in Figure 79, a sharp rise at about 1.00 \$ total uncompensated reactivity, followed by a much slower, nearly linear, rise.

For transients started at normal operating power, 40 MW, with ramp inputs, the power simply increases in a nearly linear fashion, with the strong negative reflector temperature effect damping out any tendency to an exponential rise.

7.3 Reactivity Accidents

7.3.1 Improper Rod Withdrawal. We consider here the results of withdrawing the control rods at the maximum possible rate. The rod system allows motion of either a single rod or of a gang of eight rods at 4 in./min. The ganged withdrawal is clearly the case of maximum reactivity effect. The reactivity rates of both single and ganged rods are shown in Figure 80. The ganged rod reactivity injection rate is about .08 \$/sec at the critical rod position, and diminishes to about .01 \$/sec at the limit of the motion.

Since the rod speed of 4 in./min has been deliberately chosen to be slow enough so that the neutron safety channels and safety rod drop will detect the transient and terminate it promptly there is no danger to the reactor system. The startup accident has been examined in a series of analog computer calculations for various initial powers and assumptions about the safety channel response and rod drop (14). Figures 81 and 82 show the power behavior for initial powers of 1 KW (about the normal startup level) and .01 watt at a constant ramp input of .10 \$/sec, which is about twice the actual, average ramp rate for the withdrawal of an eight rod gang. The .01 watt level is a lower limit to the neutron level in a clean, cold system with no other source than photoneutron production in the D₂O from the radioactive shield walls around the vessel. In these calculations only the high level safety channel trips are assumed to operate. In an actual case, the period circuits in the safety channels would lead to a trip at a much lower power level. Further, in the calculations, the safety rod drop is assumed to take place from the least favorable position, i.e., the fully withdrawn position. As may be seen in Figures 81 and 82, even with these assumptions the shutdown is effective in limiting the peak power to acceptable levels. The maximum fuel temperature in these cases is below 250°C.

7.3.2 Beam Tube or Thimble Collapse. The physical change considered here is a massive leak or total collapse of a single beam tube or irradiation thimble, with subsequent filling by D_2O . The critical experiments have shown that the greatest reactivity change from such an event would follow flooding of a central irradiation tube (14). The added reactivity would be 0.39% k.

The maximum effect of such an accident would be felt for an initial power of 40 MW. The power behavior of the reactor in this accident is indicated in Figure 83, where a step reactivity input of 0.5 % k is imposed on an initial power of 40 MW. The shutdown is based upon a high level trip at 60 MW from the neutron safety channels, with no period input, and a subsequent drop of only three of the eight safety rods from the full-out position. The power peak is less than 80 MW, and the maximum fuel temperature does not exceed 250°C.

It is concluded that the gain in reactivity caused by rapid flooding of any of the re-entrant beam tubes or thimbles will not endanger the reactor system. It is worth noting that the central thimbles are normally D_2O -filled for sample cooling, although future experimental programs may require them to be voided (the beam tubes are normally empty).

Further, the discussion in Sections 7.2.3 and 7.2.4 shows that for this magnitude of reactivity step input, the power level simply rises to a new equilibrium value of about 80 MW, with maximum fuel temperatures below 250°C, even with no safety rod action (see Figure 76).

7.3.3 Broken Control Rod. We consider here the results of a fracture in one of the control rod blades or followers. Since the shutdown reactivity worth of the 16 rods is some 39 % k, it is apparent that this accident will cause a power transient only when the rods have been withdrawn to the critical position. The maximum reactivity in a single rod at critical will occur for a clean, cold core configuration, and will be $10.7 \% / 16 = 0.67 \% k$.

The fracture of an auxiliary rod follower will allow the rod to drop to the full-out position, while a fracture of the blade at the attachment to the follower might allow the blade to drop to the bottom of the tank. Since the auxiliary rod is shadowed by the lattice plate in the full-out position, the reactivity effect is the same in both cases, 0.67 % k.

The fracture of a main rod follower allows the rod to drop to the fully-inserted position, where the follower-to-blade coupling block will be held by the transition plate. The effect in this case is a negative reactivity input of 4.1 % k, which promptly shuts the reactor down. A fracture of the main rod blade below the follower attachment, however, will allow the main rod blade to fall past the core to the bottom of the vessel. In this case there is first a strong negative reactivity pulse of 4-5 % k followed by a return to critical and subsequent reactivity input of +0.67 % k.

In all of these cases the power transient which is induced may be conservatively approximated by a step input of 0.67 % k at an initial power of 40 MW. (The actual single rod worth at 40 MW, for a clean core, is 0.60 %, since the temperature reactivity loss is 1.2 %.)

The power behavior of the reactor will be the same as that shown in Figure 83, with a power peak of about 90 MW, and maximum fuel temperatures below 250°C, even for the conservative assumptions of no period trip and only three of eight safety rods dropping from the full-out position. A rod drop from the critical position which gives the maximum step input used here will terminate the power pulse sooner, and with a lower peak power.

It is concluded that the fracture of a control rod, either main or auxiliary, will not endanger the reactor system.

Further, an interpolation between the step transients presented in Section 7.2.4 shows that even if no safety rods drop, the power simply peaks at about 150 MW and falls rapidly to a new equilibrium level of about 110 MW, with maximum fuel temperatures during the transient of less than 350°C.

7.3.4 Light Water Contamination. The effects of H₂O contamination on reactivity have been discussed in Section 3.3.8. It was pointed out that at H₂O contaminations below 20% there is a net reactivity loss from this source; beyond this point there is a reactivity gain which at complete H₂O is about 10 % k.

The possible sources of H_2O inleakage to the reactor primary system are the three heat exchangers EA101A, EA101B, and EA103, and the reactor vessel itself. Since the primary system is maintained at a higher pressure than the secondary (150-200 psi against 40 psi), heat exchanger leaks will be from the D_2O side into the cooling water. The presence of such leaks is determined by the radiation monitors on the cooling water effluent pipes of the exchangers. The monitors detect nitrogen and sodium activity in the heavy water.

If heat exchanger leaks occur which are of a sufficient size to make the heavy water loss undesirable, the offending unit may be valved off on both primary and secondary sides and drained to storage tanks. Since no perceptible amount of H_2O will leak into the primary system from this source, it is concluded that heat exchanger leakage presents no hazard to the safe operation of the reactor.

There remains the possibility of H_2O introduction into the reactor vessel during fuel changing operations when the tank is open. The expansion space at the top of the vessel has a volume of about 30 ft³, and during most of the time the vessel is open, this is decreased to 8 ft³. The maximum available volume for H_2O addition then will hold 224 gallons, and the reduced volume only 60 gallons. The primary system contains 7800 gallons and the reactor vessel holds a total of 2400 gallons, so that the possible H_2O addition is 2.9 % of the primary, or 9.4 % of the vessel water at the most. Additions of even the maximum amount of H_2O without overflowing the system will then result in a decrease in reactivity rather than an increase.

Of course, if a hose is turned into the open vessel and H_2O allowed to run in indefinitely, the primary fluid will eventually be nearly 100 % H_2O . It has been pointed out that for complete H_2O flooding, the control rods may not have sufficient shutdown margin to keep the reactor subcritical (38). We are unable, at this time, to say definitely whether this is the case or not. In any event, a large tank of liquid neutron poison solution which may be forced into the reactor vessel is kept in readiness on the Operations Level, as described in Section 3.8.8. This poison solution, when injected into the primary system, provides an ultimate shutdown mechanism for the reactor in either light or heavy water.

The poison solution may also be injected into the experimental D₂O system, so that the in-core and core-edge irradiation thimbles become strongly absorbing structures. The in-core thimbles, when filled with the neutron absorbing solution, should be particularly effective in controlling the core in the light water flooding case, since they will be the region of maximum poison worth for this case.

To reduce the chance of light water getting into the reactor vessel during fuel handling operations, care has been taken in the design to keep sources of light water away from the pile top. There are no light water supply lines, sprinkler pipes, or fire hose connections on or around the pile top. Within the reactor pit there are light water manifolds for the thermal shield and shutter cooling. This system is closed, however, and has a total inventory of light water which is much less than the 1500 gallons needed to reach 20 % H₂O in the primary system.

Most of the sources of water inleakage to the reactor vessel from the reactor pit and pile top are D₂O systems supplying the various experimental thimbles. There is also the emergency cooling line for fuel handling which carries D₂O. Aside from the thermal shield cooling water in the pit mentioned above, the only possible light water source around the reactor vessel would be a cooling system for the control rod mechanisms. The need for this latter system has not been established as yet, and will depend on the type of drive mechanism selected.

7.3.5 Cold Water Accident. A consequence of the negative temperature coefficient of the reactor is the possibility of cold water entering the core and reflector and causing a transient. The relatively large negative coefficient of the HFBR makes this accident an important one to consider.

The only source of cold D₂O for this accident is the inventory on the primary sides of the main heat exchangers. Each exchanger contains about 1500 gallons of D₂O, more than enough to fill the spherical section of the reactor vessel.

The accident requires that one of the main heat exchanger loops be shut down, since circulation through the exchangers will maintain the whole primary system at nearly the same temperature (the D₂O temperature change in the core at the full 40 MW power is only 7.5°C). If both heat exchanger loops are shut down, then the control rods cannot be withdrawn, and the very large reactivity holddown in the rod array makes the cold water accident impossible. Nor can the rods be withdrawn to the critical position quickly enough after the main pumps are started to make the accident possible. It requires an absolute minimum time of 5-1/2 minutes to withdraw the rods to the critical position and in this time the primary water has made 12 complete passes through the system. The only available configuration of the system which can lead to the cold water accident, then, is one in which a single loop is operating at normal flow and the other loop is shut down with cold D₂O in the exchanger. Starting the second pump will add cold D₂O to the stream, and if the temperature differences are large, will make the cold water accident possible.

The normal operating procedure for starting a main pump requires that the discharge be throttled initially, and slowly opened as the pump reaches normal speed. The procedure will insure a slow mixing of the cold D₂O in the exchanger with the circulating D₂O in the operating loop. For the purposes of this analysis, however, we shall assume an instantaneous start of the cold loop to full flow.

The maximum cold water accident will occur with a low reactor power, so that the inlet water is not heated as it passes into the reflector, and with the largest possible temperature difference. We shall assume, therefore, that the reactor is operating at low power, but with the circulating water temperature at the full power core outlet value of 56°C. The secondary water flow through the operating heat exchanger may be throttled to achieve this condition. The shut down main exchanger is assumed to have a large enough secondary water flow so that the contained D₂O reaches the secondary temperature, which will be taken as 5°C, one degree above the freezing point.

The second loop is now assumed to be activated, and to come instantaneously to full flow. The resulting slug of cold water is an equal mixture of the streams from the two loops. Its temperature will be 30.5°C , a temperature drop of 25.5°C from the operating point. The reactivity effect will be an addition of $0.245\% \text{ k}$ in 50 msec as the cold slug passes through the core, and an addition of $0.722\% \text{ k}$ in 4.1 sec as the reflector is cooled. The total reactivity input is $0.967\% \text{ k}$.

Since the reflector effect is relatively slow, the neutron safety channels and the subsequent safety rod drop will respond essentially to the $0.245\% \text{ k}$ step from the core. The power behavior will resemble that of Figure 83, but with a peak power closer to the 60 MW trip level since the input is smaller and the initial power is lower. By the time the reflector reactivity effect becomes appreciable, the reactor is safely shut down by some $39\% \text{ k}$.

It is concluded that the cold water accident offers no danger to the reactor even under the artificially severe assumptions made above.

The combination of the initial step input followed by the cooling of the reflector is an interesting transient case and calculations have been made to determine the reactor behavior with no safety rod action. The reactivity inputs are those above; a step of $0.245\% \text{ k}$ followed by a ramp of $0.176\% \text{ k/sec}$ for 4.1 seconds. The initial condition power and uniform temperature are 1 MW and 56°C , and the control rods are fixed in the initial position. The steam void fraction, f_{ag} , is taken to be .002. The calculational methods of Section 7.2 are used. The results are shown in Figure 84. The power peaks at 135 MW, the maximum fuel temperature at 308°C , and the reflector has a minimum temperature of 35°C . As the power rises, the water passing into the reflector is heated, and eventually this effect raises the reflector temperature enough to damp out the power transient.

7.3.6 Fuel Handling Criticality. The core of the HFBR is a region of poor neutron moderation. Most of the neutron slowing down takes place in the reflector region. It appears natural that any improvement in neutron moderation in the core, which would reduce the mean energy of neutrons before they escape from the core would improve the reactivity. This is, in fact, the case and it has important effects on the fuel handling procedure.

As fuel elements are removed from the center of the core, the reactivity rises. The critical experiments have shown that the maximum reactivity would occur when fourteen fuel elements are loaded in the peripheral fuel positions, and three others remain in the center region. Furthermore, the experiments have shown that full insertion of any four of the eight safety (main) rods will reduce the reactivity of this core configuration to below the level of criticality. Figure 85 shows some of the results of these studies.

The configuration of maximum reactivity will occur in the course of fuel transfer from the inner to the outer core regions and with some new fuel inserted in the inner region. However, fuel will not be moved unless all safety rods and all auxiliary rods are fully inserted. With this large hold-down of reactivity in place, the core is subcritical by nearly 20 % in k even at the maximum reactivity configuration.

It is realized that the loading of a reactor is commonly done with some safety rods partially withdrawn. However, the arrangement of interlocks in the safety system will preclude this practice for the HFBR because the control rod clutches cannot be activated unless substantial flow exists in the primary coolant circuit. It is believed that this requirement should not be compromised.

The safety rods provide a very large safety margin. Furthermore, there are no extra fuel positions in the reactor, so that overloading cannot occur. Since there is no way by which the reactor could be made critical with all rods fully inserted, it is believed that the safest loading configuration for this reactor is that suggested above, with all safety and auxiliary rods inserted.

It is concluded that there could be no criticality accident in the reactor during fuel handling if this procedure is followed.

7.4 Equipment Failures and Malfunctions

7.4.1 Fuel Cladding Failure and Plugged Water Channel. These items are discussed together because they both lead to the distribution of fission products in the primary system. In the event of a cladding failure the exposed fuel alloy will yield fission products to the primary water. If the cladding failure is in the nature of a blister rather than the usual pin hole, and if the affected portion of the fuel plate is in a high power density region, local fuel melting is to be expected. In the plugged water channel accident, due presumably to a foreign object lodged on top of the channel, the flow starvation in the channel will lead to melting of the affected area.

In each of these cases the presence of excessive amounts of fission products in the primary water will be signaled by the fission product monitors and the reactor will be shut down. The volatile fission products, notably iodine and the rare gases, will tend to collect in the helium gas space in the top of the reactor vessel. They are purged through the charcoal and particulate filter FD-105 to the exhaust gas system, filtered again, and discharged up the stack. The resulting doses calculated below are only a few per cent of the maximum credible incident doses detailed in a later section. Further, the purge rate from the helium surge volume is controlled, and it is possible to hold up the purge or reduce the flow rate until some of the radioactivity has decayed. The nonvolatile fission products are removed by the D₂O filters and ion beds in an extended period of circulation following the meltdown. The filters and ion beds have been sized to remove a complete fuel element without regeneration, so that there is ample resin capacity.

To assess the magnitude of the radioactivity hazard from the accident described above, consider the melting of 3 in.² of fuel plate at the hot spot. The volume of fuel alloy melted will be 1 cm³. The fission product inventory of this fuel alloy at the high power density point in the core will be taken at the end of a 40 day operating period, and at 100 sec after the accident. The fuel alloy volume per element is 261 cm³, and the peak-to-average power ratio is 2.82. Thus, the melted alloy has a fission product inventory at 100 sec decay equal to the average element values in Table XXIV times the factor

$$\frac{2.82}{261} = .0108$$

The values are:

Halogens	4.22×10^3 C
Rare gases	4.15×10^3
Solids	2.63×10^4
TOTAL	3.47×10^4 C

The experiments of Creek, Martin, and Parker (39) show that the release of halogens from molten aluminum fuel alloy at 700°C is 5%. The rare gas release in the same circumstances is 15%. Thus, the release of volatile fission products into the gas blanket of the reactor vessel is 2.11×10^2 C halogens and 6.23×10^2 C rare gases. The solids stay within the primary system and are removed in the D₂O ion exchange beds.

Before release to the stack the fission product gases are filtered through the charcoal filter, FD-105, and the silver-plated copper mesh exhaust filter. The rare gases pass through both of these filters but the halogens are reduced by a factor of 50 in the charcoal filter and a subsequent factor of 20 in the silver-plated copper mesh filter. The discharge of volatile fission products from the stack is then

Halogens	2.11×10^{-1} C
Rare gases	6.23×10^2

in the case where the fission products are discharged to the stack without any hold-up time. The conditions of discharge from the reactor stack are the same as those in Section 7.6 below and cloud centerline doses may be inferred by comparison with the analysis given there. The maximum integrated doses are of the order of 10^{-4} AED, and are less than 10 mR.

The limited meltdown accident described here is not expected to result in above tolerance radiation levels within the building. It is worth noting that most of the high power density water reactors now in operation have experienced at least pin-hole cladding breaks, and that in all cases the cleanup operations were carried out successfully and the reactors returned to operation.

7.4.2 Primary System Rupture. Considered here are the consequences of breaks in the primary circuit, either in the core vessel or in the piping or equipment. Since the primary circuit is a low temperature, low pressure (150°F, 275 psi), relatively unstressed piping system, such breaks are not likely. Further, since the primary coolant is well below the

atmospheric boiling temperature, a system break leads to none of the steam pressure problems attendant upon a break in a water power reactor system.

The system breaks may be considered in two classes: those below the outlet nozzle in the reactor vessel itself, and all others. In the event of a rupture of the vessel below the outlet nozzle, D_2O fills the thermal shield vessel and the beam tubes and shutters inboard of the seal caps. The thermal shield vessel fills to the top and the core remains immersed at all times. Since several process instruments will trip the safety rods in this accident (vessel pressure, liquid level, core pressure drop, primary flow) the fission power is shut off abruptly when the break occurs. The afterheat is dissipated to the D_2O in the thermal shield cavity by boiling and the natural circulation set up in this case by the opening of the primary bypass valves, as described in Section 3.8.7. Since the remaining D_2O in the primary system may be drained to the storage tank, FA-101, and returned to the core region via the emergency makeup system, an ample supply of makeup water is available.

System breaks occurring at any other part of the primary circuit lead to the same conditions but with D_2O retained in the spherical section of the reactor vessel. As noted in Section 3.8.6, a siphon-break bypass is provided between the inlet and outlet nozzles on the reactor vessel. The primary system arrangement, then, insures that the core is covered with D_2O in all cases.

Radioactivity hazards to be considered in the primary system rupture accidents are limited to the problem of tritium in the air exhaust, since the core remains intact, and the spilled D_2O is retained within the primary containment shields. The air exhaust rate from the primary spaces is about 200 cfm and must be maintained to insure a negative pressure in those spaces. After a major spill, this air will be saturated with D_2O vapor, and after the natural circulation boiling starts in the core, the air carries some low temperature steam as well. It is expected that much of this steam will condense in the exhaust ducts, and most of the remainder in the filter system.

To assess the magnitude of the hazard we assume that all of the D₂O steam formed in the natural circulation boiling is carried through the exhaust system to the reactor stack. The steam rate will be about 1500 lbs/hr, diminishing as the afterheat decays away. The tritium content after many years of reactor operation at 40 MW will approach $1.80 \times 10^4 \mu\text{C}/\text{cm}^3$, or 7.43 C/lb D₂O (35).

The tritium release rate is then

$$\frac{7.43 \times 1.5 \times 10^3}{3.6 \times 10^3} = 3.10 \text{ C tritium/sec}$$

The stack exhaust flow in normal operation is 134 m³/sec. The tritium concentration in the stack exhaust air stream is then

$$3.10/134 = .0231 \text{ C/m}^3$$

The atmospheric dilution of the tritium concentration for points at the site boundary and beyond will be a factor of about 10⁵, which will reduce the tritium concentration to .23 $\mu\text{C}/\text{m}^3$, , , about 1/30th of the tolerance value even if the release is continued indefinitely.

For short-time exposures within several hundred meters of the stack, it has been found that in strong lapse conditions with looping of the stack effluent plume, the dilution factor from stack top to the ground may be as small as 5000 (40). In these conditions the ground level concentration of tritium in this accident case may momentarily reach values as high as 4.6 $\mu\text{C}/\text{m}^3$. Even this concentration is well below the occupational tolerance of 20 $\mu\text{C}/\text{m}^3$ (36).

7.4.3 Power Failure. Loss of power from the normal commercial lines may occur several times a year, usually for a duration of only a few seconds. It is important that the reactor system be able to function safely even in a prolonged power outage, however. As noted in Section 4.3.2, an emergency electrical system of storage batteries plus a propane engine generator set will supply the needed shutdown condition power for the HFBR for an indefinite period.

Since the safety instrument channels and the control rod clutches themselves are energized from the normal a.c. lines, failure in this electrical supply results in an immediate reactor shutdown. Once the fission process is stopped, the essential functions are the afterheat removal from the core and maintenance of the building air exhaust. The emergency electrical system supplies power for these functions, and for lights and some instruments.

If the emergency power source should fail, the bypass valves in the vessel open (a non-electrical system) and the afterheat is dissipated in natural circulation boiling, as described in Section 3.8.7. Since the primary system will not be open in this case, D₂O steam is condensed in the shutdown heat exchanger and there is no discharge of tritium-containing vapor to the air exhaust. The building air intake damper closes upon failure of the emergency power, and the building is maintained at a negative pressure by the natural draft of the stack.

7.4.4 Loss of Pressure. At full power the reactor primary system is pressurized to insure that there is no local boiling at the hot spots in the core. This pressurization is accomplished by introducing high pressure helium into the top of the reactor vessel. A rupture of the helium recirculation piping, or of the vessel cover, would relieve the helium pressure in a relatively short time, and the consequences of these events must be examined for possible hazards.

The loss of pressure will be detected immediately by the pressure instruments, and the safety rods will act to shut down the reactor. The core, however, will protect itself by a self-induced power reduction whether the rods act or not. As soon as the system pressure drops to atmospheric, steam void formation in the core causes a reduction in the power. No fuel is melted and there is no escape of fission product activity. A minor amount of tritium is carried into the building in the blowdown of the helium pressure, but the amount is small compared to that in the primary system rupture case of Section 7.4.2, and leads to doses well below tolerance outside the building.

The pressure loss transient has been calculated by the methods described in Section 7.2. The reactor is assumed to be operating at 40 MW with the normal power and temperature distribution when the pressure is abruptly reduced to atmospheric in the blanket gas space. No control rod motion is assumed. The results of this calculation are shown in Figure 86. Both the power and maximum fuel temperatures drop to new equilibrium values in less than .4 sec. A calculation of the steady state burnout heat flux for the new conditions gives 1.53×10^6 BTU/hr-ft², compared to the new hot spot heat flux at 37.1 MW of 1.01×10^6 BTU/hr-ft².

It is concluded that the loss of pressure, even at full power, does not endanger the reactor system.

7.4.5 Instrument and Emergency Bypass Malfunctions. All of the critical instrument points are covered by two or three parallel units. Thus, there are three neutron safety channels, three system pressure detectors, three reactor vessel liquid level instruments, and two core pressure drop detectors. The failure or malfunction of an instrument, then, does not leave the reactor system unprotected.

The reactor is instrumented so that certain safety measures are taken automatically in the event of a loss of flow to the core. These actions are described in detail in Section 3.8.7. In brief, they are the opening of the emergency bypass valves in the reactor vessel transition plate, the opening of the siphon break bypass valve VB101, and the opening of the reactor vessel relief valve SV101B (see Figure 34). These actions occur either upon the opening of all four pump motor breakers, GA101A, GA101B, GA102A, and GA102B, or upon a "low-low" liquid level signal from the instruments LA103 and LA102 (see Figure 34). In the case of a loss of flow to the core these various measures must be taken promptly to insure the removal of the fission product afterheat from the core.

At the same time, there is a possibility that the various bypass and relief valves may be caused to open during normal operation by a malfunction of the instrument system. We shall consider here, then, an accident in which these valves open while the reactor is operating at normal flow and full power, with the normal power and temperature distribution.

This "bypass accident" resembles the loss of pressure case of Section 7.4.4, with the additional effects of the opening of the siphon break and emergency bypass valves to be considered. The gas blanket pressure is dropped to atmospheric, as in the pressure loss accident. The normal flow of the heavy water continues, but about 65 per cent of the flow is bypassed around the core through the emergency bypass valves. Another 5 per cent bypasses the core through the siphon break. The water velocity in the fuel element coolant channels drops from the normal 35 ft/sec to 10 ft/sec.

The power and fuel temperature behavior in this accident are the same as that shown in Figure 86 for the loss of pressure alone. The power reduction is greater, however, because the lower water velocity allows more steam void to be effective in the core. A calculation of the steady state burnout heat flux for the reduced pressure and water velocity gives a value of 1.02×10^6 BTU/hr-ft². The hot spot heat flux at the new equilibrium power of 30 MW is 0.82×10^6 BTU/hr-ft². We conclude that the malfunction of the bypass valves does not endanger the reactor system.

7.5 Fire and Environmental Hazards

7.5.1 Fire. The reactor building and internal structures are of concrete and steel for the most part, and are, therefore, not flammable. Limited area fires are possible in overloaded electrical circuits, and at breaks in the propane gas lines which serve the emergency generator set. Such fires will not endanger the safe functioning of the reactor safety mechanisms.

7.5.2 Earthquakes. As noted in Section 2.5, the likelihood of serious earth tremors in the BNL area is small, and the building and reactor structures are designed for horizontal accelerations of 98 cm/sec², as compared to maximum recorded accelerations of about 8 cm/sec². Damage to the reactor or building from seismic activity is, therefore, not expected.

7.5.3 Flood. Long Island is not subject to general floods, except near the seashore during storms. The Laboratory is approximately in the center of the Island, about 6 miles from Long Island Sound to the north and an equal distance from Great South Bay on the south. The general elevation of the Laboratory is some 70-90 ft above sea level, so that sea water flooding is not a hazard. The BNL area terrain is mostly flat, with a few low hills and no large rivers, so that fresh water flooding is similarly unlikely.

7.5.4 Hurricane. As noted in Section 2.3, the area is subject to hurricanes, with wind velocities which approach 120 mph. The reactor building and other reactor structures nearby are designed to withstand these wind loads with an ample safety factor.

7.5.5 Interaction with the Graphite Reactor. The proximity of the Graphite Reactor to the HFBR, and their sharing of some facilities, makes it necessary to consider how, if at all, either could contribute to a reduced safety of the other.

The easiest interaction to dispose of is that which might occur through exchange of neutrons. The primary requirement on shielding of both systems is the reduction of personnel exposures to tolerable levels. The two shields together attenuate neutrons by a factor greater than 10^{20} . Since the total neutron leakage from each reactor is about 10^{18} neutrons/sec, and since the separative distance is about 300 meters, it is apparent that the number of neutrons entering one core from the other is truly microscopic.

Since it is proposed that the HFBR exhaust air be discharged into the stack of the Graphite Reactor, careful consideration has been given to the possibility of exhaust gases from either reactor being forced into the other via the exhaust ducts. Tests have been conducted on the stack which prove that under no conditions does the negative pressure at the stack base go to zero (33). The least negative pressure observed in the tests was -0.16 in. H_2O , and this value has been used to determine the minimum exhaust conditions in the HFBR building. The stack is of such large capacity (27 ft base I.D., 17 ft top I.D., 350 ft above Laboratory ground level) that a negative pressure will be maintained at its base notwithstanding extreme fluctuations in the amount of exhaust air supplied from either reactor. Accordingly, there can be no flow from one reactor to the other.

Should an accident in the HFBR liberate activity into the building air system, the ordinary stack circulation pattern would assure exhaust in the normal way rather than backup into the Graphite Reactor air ducts. This situation would prevail even during a complete Laboratory power failure; in this case the natural convection afforded by the stack would maintain flow in both systems.

It should be noted that the tendency of the stack to produce useful draft even on hot days is amplified by 1) the heating of the air by the Graphite Reactor fans if they are operating, or 2) the heat stored in the graphite. The heat stored in the graphite is available even if all electrical power has been lost, and the stack draft is due only to natural convection. Any wind which might be present also supplies an aspirator effect.

7.6 Maximum Credible Incident

7.6.1 Basis for Choice. The considerations outlined up to this point have shown that there is no credible way to cause release of substantial amounts of fission product activity from the core to the building atmosphere. The only believable releases of this kind might be the result of local channel blockage. In this case, the region from which release took place would be quite localized, and only a small fraction of the core fission product inventory would be involved. The fission products would be completely contained in the primary reactor circuit until they were purged out in the exhaust gas system or removed by the demineralizer system for disposal by normal methods.

There is, however, a possibility of fission product release at another time, viz.: when fuel handling is taking place. The fuel handling sequence involves lifting an element from the core vessel and transferring it to a chute which leads to the canal. When an element is being removed and is still under water, natural convection will suffice to carry away the afterheat. After the removed element has fallen through the fuel chute into the canal, natural cooling will again take care of the situation. During the transfer, however, artificial cooling must be supplied if the element is to be out of water for longer than a few minutes. This is provided by a forced circulation air system which passes 200 cfm of air through the element during the transfer from the primary vessel to the chute. Should this air cooling system fail, a D₂O spray onto the element at approximately 1 gpm will prevent melting.

It is not inconceivable, however, that both cooling systems will fail together, in which case the fission product heat generation will be great enough to melt the element. It is this melting and the associated fission product release which we consider to be the maximum credible incident.

7.6.2 Description of the Incident. The maximum credible incident is assumed to take place in the course of changing the fuel 24 hours after a normal 20-day operating period. At this point the outside 14 fuel elements are to be removed, since they have been in the reactor for two half-cycles, and have experienced the full 20% burnup. The inner 14 elements are then to be moved to outside positions, after which the inner 14 vacancies are to be loaded with fresh fuel.

It is assumed that one of the elements to be removed from the reactor is in transit between the core vessel and the fuel chute to the canal. At some point in the transfer, a power failure occurs, so that the full 200 cfm of cooling air is no longer applied. Furthermore, both the transfer release mechanism and the emergency D₂O spray cooling are assumed to fail. The fission product afterheat then causes the element to melt in about 10 minutes. The melt falls on metal surfaces and solidifies, or falls into the reactor vessel or the fuel chute. During the molten phase, fission products are released.

The important components, in light of what may be released from an aluminum melt, are the halogens and the rare gases. We deduce inventories of these in the element at 24 hours after reactor shutdown from Table XXIV, with a decay time dependence of $t^{-.21}$ (a factor of 0.241 for the 100 sec to 24 hr decay):

Halogens	$1.33 \times 10^5 \text{ c}$
Rare gases	$1.32 \times 10^5 \text{ c}$

From the experiments of Creek, Martin, and Parker (39), the release from the molten aluminum fuel is 5% of the contained iodines and 15% of the contained rare gases. We double these release fractions to account for the differences in melting conditions between the accident considered and the Oak Ridge experiments. The assumed output from the molten fuel is then:

Halogens	$1.33 \times 10^4 \text{ c}$
Rare gases	$3.96 \times 10^4 \text{ c}$

The exhaust air flow from the reactor pit (which is not affected by the power failure) will remove the bulk of the fission product gases in about 100 seconds. On the way to the stack, the halogen inventory is reduced by a factor of 20

in the passage of the gas through the absolute and silver-plated copper mesh filters. The release at the stack top will then consist of:

Halogens	665 C
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Rare gases	39,600 C
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this release being spread out over an interval of about 100 sec.

7.6.3 Results of the Accident. An inspection of the circumstances of release shows that meteorological conditions prevailing during the day would lead to the worst personnel exposures. The unusually high release point set by the stack height (350 ft above ground) combined with the low rate of vertical mixing in nocturnal inversion conditions causes a cloud released at night to travel for long distances before it has approached the ground. For this reason, the daytime diffusion conditions produce the higher ground level concentrations.

In estimating the worst daytime doses, it is necessary to assume diffusion parameters which are different from those used in most previous studies at this Site. The reason for this is the relatively short release time. In this case the larger scale horizontal motions play no part in diffusing the material, but merely affect the location of the plume. The calculation of diffusion has been based on the standard Sutton equation (41), and the parameters chosen are $C_y = 0.15$, $C_z = .40$, $n = 0.25$, and $u = 3.0$ m/sec. It is assumed that the plume is released at the 108 meter stack height with no vertical rise whatever.

The results of the calculation are shown in Figure 87. The peak of the centerline concentration occurs at 589 meters from the release point; the integrated cloud concentrations at this point are:

Halogens	1.19×10^{-2} C-sec/m ³
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Rare gases	7.09×10^{-1} C-sec/m ³
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By the methods of WASH-740 (42), the following approximate personnel doses at this point would follow, in terms of the Acceptable Emergency Dose defined in that report:

	<u>Halogens</u>	<u>Rare Gases</u>
Thyroid	0.000972 AED	---
Whole body γ	0.000595	0.0342 AED
Lung β	<u>0.000853</u>	<u>---</u>
Total	0.00242 AED	0.0342 AED

The total maximum exposure, then, is about 0.037 AED at the point of greatest ground cloud concentration. Since the maximum dose is less than the AED, the exclusion area, low population zone distance, and population center distance, as defined in the Reactor Site Criteria Guide, are not applicable.

7.7 Supplementary Safety Considerations

The recently published Reactor Site Criteria Guide commends to the attention of the reactor designer ". . . a more serious accident (not normally considered credible) . . ." in addition to the usual hazards analysis which has been detailed in the preceding sections. It is against this ultimate eventuality that the reactor building shell has been redesigned (from the preliminary design presented in the Site Report) to be gas-tight to a high degree, and to withstand a very large energy release, reflected as an internal gas over-pressure. In a similar vein, the reactor primary shielding has been strengthened with steel reinforcing and hold-down members, and has been encased on the outside surfaces with heavy steel plates.

To assess the magnitude of the hazards involved in such an accident a fission product release to the building atmosphere identical to that in Appendix A of the Site Criteria Guide is assumed (see Table XXIV):

100% of the rare gases	= 9.47×10^6 C	(at 100 sec after shutdown)
50% of the halogens	= 4.85×10^6 C	
1% of the solids	= 7.05×10^5 C	

Of this release, one-half of the halogens are assumed to adsorb onto internal surfaces of the building or to internal components.

The release of the building atmosphere radioactivity to the environment is assumed to occur at the specified leak rate for the building at the design overpressure. For the HFBR building this is 1% of the contained volume per day at 2 psig. An energy release of 1800 MWS is required to heat the building atmosphere to 143°F, to cause a 2 psig overpressure. Since the hot atmosphere will transfer this energy to the internal structures, as well as to the environment through the building shell in a few hours, the overpressure condition will be terminated in 8 hours. During the overpressure period, it is assumed that the leak rate is constant at the full 2 psig value.

On the basis of these release values, and assumptions pertinent to the Brookhave site and meteorology, calculations of the exclusion area, low population zone and population center distances have been made in accordance with the suggestions contained in the Reactor Site Criteria Guide.

In addition to the release rates already mentioned, certain circumstances have been assumed. In the first place, there is no valid reason to believe that such a release, if it occurred at all, would not occur at the least favorable location, at or very near ground level. Secondly, the temperature inversions are the least favorable meteorological conditions for a ground-level release, and they are quite common at this site. For all computations, therefore, the following values for the Sutton diffusion approximation have been used:

$$h = 0$$

$$n = 0.50$$

$$C_y = 0.40 \text{ m}^{n/2}$$

$$C_z = 0.07 \text{ m}^{n/2}$$

$$\bar{u} = 1 \text{ m/sec,}$$

and the wind direction has been assumed to be invariant.

A rough calculation of the cloud thyroid and whole body doses showed that the thyroid dose would be the controlling factor in setting the exclusion area and low population distance. The cloud concentrations of the individual iodine isotopes were therefore calculated and integrated over the various times of interest to obtain the cloud dose in curie-

seconds/meter³ for each isotope. The conversion to Rem was done according to the values given in the Site Guide, and the total iodine thyroid dose in Rem obtained as a function of distance and exposure time.

The iodine inventory for the five isotopes of interest is given in Table XXV.

The conversion table from C-sec/m³ to Rem for the five iodine isotopes is given in Table XXVI.

Table XXV

Iodine Isotope Properties for Dose Calculations

	<u>I¹³¹</u>	<u>I¹³²</u>	<u>I¹³³</u>	<u>I¹³⁴</u>	<u>I¹³⁵</u>
In core, 100 sec after a 40 MW, 40-day run: units of 10 ⁶ curies	.87	1.46	2.1	2.77	1.89
In building, available for leakage, 100 sec after a 40 MW, 40-day run: units of 10 ⁶ curies	.218	.365	.525	.693	.473
Leak rate at 1%/day, 100 sec after a 40 MW, 40- day run: curies/sec	.0251	.0422	.0607	.0800	.0547
Decay constant: sec ⁻¹	.996	.802	.925	2.20	2.89
	x 10 ⁻⁶	x 10 ⁻⁴	x 10 ⁻⁵	x 10 ⁻⁴	x 10 ⁻⁵

The results of the calculation are shown in Figure 88. It should be noted that the longer exposure times, 10 and 100 days are calculated on the assumption that the unfavorable diffusion conditions of the inversion continue throughout the exposure period. Such an assumption is clearly unwarranted, since inversions normally dissipate within 12 hours. A more realistic dose calculation for the long exposure times would show that both the basic wind direction and the diffusion conditions will vary widely, and that the integrated dose at any particular point will be reduced to about the level of the 1 day case shown in Figure 88.

Table XXVI

Conversion Factors from C-sec/m³ to Rem
for the Iodine Isotopes*

<u>Isotope</u>	<u>Conversion Factor</u>
I ¹³¹	329 Rem/(C-sec/m ³)
I ¹³²	12.4
I ¹³³	92.3
I ¹³⁴	5.66
I ¹³⁵	25.3

*Values from Appendix A, Reactor Site Criteria, USAEC.

The direct gamma radiation dose from the fission products within the reactor building is calculated according to the scheme set forth in the Site Guide. The equation

$$D = \frac{483 B \exp(-\mu r)}{4\pi r^2} \int_0^t t^{-0.21} dt \quad R/MW$$

is used, with r in meters, t in seconds, $\mu = 0.01 \text{ m}^{-1}$, and $B = 1 + \mu r + (\mu r)^2/3$. The results are shown in Figure 89 for various exposure times.

From Figures 88 and 89 it is seen that the cloud thyroid dose is controlling for the 2-hr exposure time. Thus, the required exclusion distance for the assumed fission product release and building leak rate is

Exclusion Distance = 310 meters.

For the 8-hr exposure time the cloud thyroid dose is again controlling, so that the required low population distance is

Low Population Distance = 750 meters.

The population center distance is then 1-1/3 times this last, or 1000 meters.

REFERENCES

1. T. Auerbach, J. Chernick, J. Juliens, G. Lellouche, W. Zinn and Associates, Preliminary Design of a High Flux, Epithermal Research Reactor for the Brookhaven National Laboratory, Proc. Second Int. Conf. Geneva, 1958, P/424, Vol. 10, p. 60, (1958).
2. P. Michael and W. Rothenstein, Preliminary Theoretical Studies of the Brookhaven Beam Research Reactor, Trans. Am. Nuc. Soc., 2, 120, (June 1959).
3. K. Downes and A. J. Court, Power, Flux and Importance Distributions in the High Flux Beam Reactor Critical Experiments, Trans. Am. Nuc. Soc., 3, 102, (June, 1960).
4. A. Court and K. Downes, Beam Tube Flux Evaluation on HFBR Critical Experiments, Trans. Am. Nuc. Soc., 3, 103, (June, 1960).
5. J. M. Hendrie, Preliminary Design of the Brookhaven High Flux Beam Reactor, Trans. Am. Nuc. Soc., 3, 48, (June, 1960), and High Flux Beam Reactor, Interim Report on Reference Design B, Brookhaven National Laboratory (June 1, 1959).
6. J. M. Hendrie and H. J. C. Kouts, Site Report on the Brookhaven High Flux Beam Research Reactor, Brookhaven National Laboratory, (July, 1960).
7. Preliminary Design Report, High Flux Beam Research Reactor, The Lummus Co., Newark, New Jersey, (August, 1960); and J. M. Hendrie and T. V. Sheehan, BNL Supplement No. 1 to Preliminary Design Report, Brookhaven High Flux Beam Research Reactor, Brookhaven National Laboratory, (October 19, 1960).
8. M. E. Smith and I. A. Singer, Diffusion, Deposition, and Rainout of the Radioactive Cloud, Appendix E, WASH-740, U.S.A.E.C., (March, 1957).
9. Wallace de Laguna, Geology and Hydrology of Brookhaven National Laboratory and Vicinity, Suffolk County, N.Y., Part I: Geology, U. S. Dept. of the Interior Geological Survey (Sept. 1958), p. 9.

10. M. A. Warren and N. J. Lusczynski, Geology and Hydrology of Brookhaven National Laboratory and Vicinity, Suffolk County, N.Y., Part II: Hydrology, U.S. Dept. of the Interior Geological Survey (April, 1958), p. 69.
11. N. A. Christensen, A Study of Proposed Use of Ground Water at the Brookhaven National Laboratory and Its Effect Upon the Water Table, Brookhaven National Laboratory (1959), p. 7.
12. Wallace de Laguna, Geology and Hydrology of Brookhaven National Laboratory and Vicinity, Suffolk County, N.Y., Part IV: Patterns of Movement and Spread of Water Borne Contamination, U.S. Dept. of the Interior Geological Survey, (May, 1958), p. 56.
13. Ibid, p. 87.
14. Brookhaven Reports on HFBR Physics:
 - a. Experimental reports by H. Kouts, K. Downes, A. Court, et al:
 - (1) HFBR Critical Assembly Report No. 1, Experiments on Core 4 (October 4, 1959).
 - (2) HFBR Critical Assembly Report No. 2, Experiments on Core 6 (November 16, 1959).
 - (3) HFBR Critical Experiment Report No. 3, Experiments on Core 7 (November 16, 1959).
 - (4) HFBR Critical Assembly Report No. 4, Experiments on Core 8 (November 19, 1959).
 - (5) HFBR Critical Assembly Report No. 5, Experiments on Core 9 (November 30, 1959).
 - (6) HFBR Critical Assembly Report No. 6, Experiments on Core 10, (December 7, 1959).
 - (7) HFBR Critical Assembly Report No. 7, Experiments on Core 11 (Part I) (December 8, 1959).
 - (8) HFBR Critical Experiment Report No. 8, Experiments on Core 11 (Part II) (February 3, 1960).
 - (9) HFBR Critical Experiment Report No. 9, Power Maps and Absolute Flux per Watt (February 11, 1960).
 - (10) HFBR Critical Experiment Report No. 10, Experiments on Core 12 (February 17, 1960).

- (11) HFBR Critical Experiment Report No. 11, Experiments on Core 11 (Part III) (February 18, 1960).
- (12) HFBR Critical Experiment Report No. 12, Experiments on Core 13 (February 19, 1960).
- (13) HFBR Critical Experiment Report No. 13, Recapitulation of Reactivities (February 26, 1960).
- (14) HFBR Critical Experiment Report No. 14, Experiments on Core 14 (Part I) (March 22, 1960).
- (15) HFBR Critical Experiment Report No. 15, Neutron Importance in the Reflector (May 18, 1960).
- (16) HFBR Critical Experiment Report No. 16, Experiments on Core 14 (Part II) (May 24, 1960).
- (17) HFBR Critical Assembly Report No. 17, Measurement of Neutron Generation Time in BBRR (May 25, 1960).
- (18) HFBR Critical Assembly Report No. 18, Experiments on Core 15 (Part I) (June 14, 1960).
- (19) HFBR Critical Assembly Report No. 19, Review of Reactivity Balance of Core J (June 15, 1960).
- (20) HFBR Critical Assembly Report No. 20, Gamma Heating Measurements on Core 15 (November 9, 1960).
- (21) HFBR Critical Assembly Report No. 21, Power Distribution Measurements on Core 15 (December 20, 1960).

b. Theoretical Reports:

- (1) J. Juliens, Study on an Experimental High Flux Intermediate Reactor, BNL internal memorandum (April 9, 1957).
- (2) John B. Sampson, Effective Calculations for the High Flux Reactor, BNL internal memorandum (September 3, 1958).
- (3) P. Michael and W. Rothenstein, Critical Volume Calculations of D₂O Reactor using GNU Code, BNL internal memorandum (December 1, 1958).
- (4) P. Michael and W. Rothenstein, Clean Spherical HIFI Calculations, BNL internal memorandum (December 3, 1958).
- (5) P. Michael and W. Rothenstein, H₂O Contamination of BBRR Reactors, BNL internal memorandum (February 18, 1959).
- (6) P. Michael and W. Rothenstein, Burnout Calculations of BBRR, BNL internal memorandum (February 19, 1959).
- (7) P. Michael and W. Rothenstein, Reference Core Size, BNL internal memorandum (February 19, 1959).

- (8) P. Michael and W. Rothenstein, BBRR Reactivity Coefficients, BNL internal memorandum (March 3, 1959).
 - (9) P. Michael and W. Rothenstein, Shim Rod Calculation, BNL internal memorandum (March 24, 1959).
 - (10) P. Michael and W. Rothenstein, BBRR Control Properties, BNL internal memorandum (June 30, 1959).
 - (11) P. Michael and W. Rothenstein, Further BBRR Calculations, BNL internal memorandum (May 28, 1959).
 - (12) P. Michael, BBRR Neutron Leakage, BNL internal memorandum (July 14, 1959).
 - (13) Keung Luke, Fission Product Buildup in HFBR, BNL internal memorandum (September 28, 1959).
 - (14) C. Sastre, Startup Accident of HFBR, BNL internal memorandum (September 2, 1969).
 - (15) Herbert Kouts, HFBR Excess Reactivity Requirements, BNL internal memorandum (January 10, 1961).
-
- 15. P. R. Tichler, Core Thermal Analysis, HFBR, BNL internal memorandum (May 5, 1960); P. R. Tichler, Revised HFBR Thermal Analysis, BNL internal memorandum (Feb. 27, 1961).
 - 16. S. Mirshak, W. S. Durant, and R. H. Towell, Heat Flux at Burnout, DP-355, E.I. duPont de Nemours & Co. (Feb., 1959).
 - 17. L. Bernath, Extension of the Method of Burnout Prediction, AECU-3901 (1958).
 - 18. R. L. Menegus, Burnout of Heating Surfaces in Water, DP-363, E.I. duPont de Nemours & Co. (March, 1959).
 - 19. P. R. Tichler, Reactor Vessel Pressure Drop Calculations, HFBR, BNL internal memorandum (June 9, 1960).
 - 20. J. M. Hendrie, Primary System Operating Pressures and Temperatures, BNL internal memorandum (June 29, 1960).
 - 21. P. R. Tichler, Gamma Heating in Control Rods, HFBR, BNL internal memorandum (January 23, 1961).
 - 22. P. R. Tichler, Flow Coastdown, HFBR Primary System, BNL internal memorandum (August 12, 1960).
 - 23. P. R. Tichler, Hot Channel Metal and Water Temperatures During Flow Coastdown, HFBR, BNL internal memorandum (March 1, 1961).

24. P. R. Tichler, Flow Distribution and Stability at Shut-down Flow Level, HFBR, BNL internal memorandum (March 2, 1961).
25. F. B. Hill, HFBR Emergency Thermal Siphon Circuit, BNL internal memorandum (August 8, 1960).
26. P. R. Tichler, Fuel Element Heat Fluxes During Flow Coast-down, HFBR, BNL internal memorandum (January 5, 1961), and superseding memorandum of the same title, January 31, 1961.
27. P. R. Tichler, Fuel Handling--Decay Heat Removal During Transit, BNL internal memorandum (November 4, 1960).
28. J. M. O'Donnell, HFBR Thermal Shield Summary of Calculations, BNL internal memorandum (December 30, 1960).
29. J. M. Hendrie, HFBR Control Rods, BNL internal memorandum (January 9, 1961).
30. J. M. O'Donnell, Stress Analysis of HFBR Pressure Vessel, BNL internal memorandum (March 9, 1960).
31. J. M. O'Donnell, HFBR Beam Tube Cooling Requirements, Ref. 11.2, BNL internal memorandum (July 13, 1960).
32. J. F. Wett, Jr., Surface Temperatures of Irradiated ORR Fuel Elements Cooled in Stagnant Air, ORNL-2892 (April 6, 1960).
33. K. C. Hoffman and T. V. Sheehan, Utilization of No. 1 Research Reactor Stack for Proposed HFBR Building Ventilation Air, memorandum, Mech. Engr. Div., BNL (July 6, 1959); M. E. Smith, Backflow from the Research Reactor Stack into HFBR, memorandum, Meteorology Div., BNL (Oct. 7, 1959); and T. V. Sheehan, Stack Report--High Flux Reactor, memorandum, Mech. Engr. Div., BNL (Oct. 12, 1959).
34. Blast Damage to Air Cleaning Devices (Filter Tests), Air Cleaning Laboratory, School of Public Health, Harvard University (November 28, 1955).
35. J. M. Hendrie, Tritium Production in the HFBR, BNL internal memorandum (May 1, 1961).

36. Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure, National Bureau of Standards Handbook 69, U.S. Dept. of Commerce (June 5, 1959).
37. G. C. Kinne, Summary of Fission Products for the HFBR 100 Seconds After Shutdown, BNL internal memorandum (April 21, 1961).
38. E. Katine, private communication.
39. G. E. Creek, W. J. Martin and G. W. Parker, Experiments on the Release of Fission Products from Molten Reactor Fuels, ORNL-2616 (July 22, 1959).
40. M. E. Smith, private communication.
41. Meteorology and Atomic Energy, AECU-3066, U.S. Weather Bureau (July, 1955).
42. Fission Product Activity in the 500,000-tkw Reactor, Appendix C, WASH-740, USAEC (March, 1957).

Figure 1. Map of the area of Long Island, N.Y. around Brookhaven National Laboratory. The inset shows the relation of BNL to the New York Metropolitan area.

Figure 2. Map of the South Site of Brookhaven National Laboratory.
The outlined sector is shown enlarged in Figure 3.

Figure 3. Map of the populated area of Brookhaven National Laboratory showing the location of the new High Flux Beam Reactor. The circles have a radii of 100, 200, 300, 500, 800, and 1100 meters.

4a.

4b.

LAPSE $T_{410} - T_{37}$ NEGATIVE
INVERSION $T_{410} - T_{37}$ ISOTHERMAL
OR POSITIVE

4c.

4d.

Figure 4. Meteorology. 4a: 37' and 355' annual wind rose.
4b: Inversion and lapse condition wind rose.
4c: BNL area terrain map.
4d: Precipitation condition wind rose.

Figure 5. North-south geologic cross section through Brookhaven National Laboratory showing the underlying strata.

Figure 6. North-south cross section through Brookhaven National Laboratory showing the movement of ground waters in the various aquifers.

Figure 7. HFBR neutron flux curves.

Figure 8. Vertical average power distribution surface in the HFBR core.

Figure 9. Vertical power density traverses in the hot channel.

Figure 10a. Experimental control rod calibration curve for a main control rod.

Figure 10b. Reactivity effects of H_2O dilution in the primary D_2O . The curves are calculated for a spherical core and reflector system.

Figure 11. Burnout correlation for the HFBR.

Figure 12. Gamma heating measurement results for the center of the core.

Figure 13. Neutron absorption curve for an HFBR control rod poison section.

Figure 14. Control rod heating from neutron capture and core gamma flux, at 40 Mw power level.

Figure 15. Fission product after heat in the hot element.

Figure 16. Fission product after heat in the HFBR core, 1-10⁴ sec. decay.

Figure 17. Fission product after heat in the HFBR core, 10^4 - 10^8 sec. decay.

Figure 18. Schematic elevation of the reactor vessel showing flow paths and internal shrouds.

Figure 19. Natural circulation atmospheric pressure burnout flow rates as a function of heat flux in vertical heated channels.

Figure 20. Type J fuel element. The fuel alloy is 30 wt. % fully enriched uranium in aluminum. The cladding is aluminum.

Figure 21. Horizontal cross-section of the HFBR vessel, showing beam tubes and core structures.

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Figure 22. Isometric view of the HFBR core structure.

Figure 23. Control rod poison section positions around the core. The main rods drop on a scram signal, while the auxiliaries are driven in. Both sets of rods are operated ganged, and the mechanisms for both are above the core.

Figure 24. HFBR control rod details.

Figure 25. Vertical section of the shield and vessel, showing the arrangement of components.

Figure 27. Horizontal section of the HFBR central structure, taken near the top of the reactor vessel.

Figure 28. Details of the reactor vessel.

Figure 29. HFBR control rod mechanism reference design.

Figure 32. Detail of a typical HFBR horizontal neutron beam tube assembly.

Figure 33. HFBR process flow schematic diagram.

Figure 34. Detailed piping and instrument flow diagram.

Figure 35. Steam, water, gas and air flow diagram.

Figure 36. Isometric view of the primary system.

Figure 37. Emergency by-pass valve and flow diagram.

Figure 38. Fast flux irradiation thimbles, with the handling cask shown mounted on the bedplate of the thimble on the left.

Figure 39. Elevations of the fuel handling apparatus.

HFBR
FUEL HANDLING ISOMETRIC

Figure 40.

Figure 41. Nuclear instrument schematic drawing.

Figure 42. Reactor safety system schematic drawing.

Figure 43. Map of the immediate site of HFBR.

Figure 45. Building elevation - east-west direction.

Figure 48. Building plan for the balcony level.

Figure 50. General ventilation and exhaust scheme for the HFBR building.

NOTE:

1. TOTAL RESISTANCE FOR CLOSED POSITION = 150 *
2. CORRESPONDING PRESS TO CLOSE = .283 PSIG = 7.9" W.G.

SECTIONAL DETAIL OF BLAST CLOSURE VALVE

Figure 52.

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POWER vs TIME SPERT STEP TRANSIENT
 $\Delta k = .0082$, $f_{as} = .005$

Figure 53.

POWER vs TIME SPERT STEP TRANSIENT
 $\Delta k = .0087$, $f_{as} = .005$

Figure 54.

POWER vs TIME SPERT STEP TRANSIENT
 $\Delta k = .0102$, $f_{as} = .002$

Figure 55.

POWER vs TIME SPERT STEP TRANSIENT

$\Delta k = .0102,$

$f_{as} = .005$

Figure 56.

POWER vs TIME SPERT STEP TRANSIENT
 $\Delta k = .0082$, $f_{as} = .005$

Figure 57.

011 2.7

POWER vs TIME SPERT STEP TRANSIENT

$\Delta k = .0087$, $\text{fas} = .005$

Figure 58.

POWER vs TIME SPERT STEP TRANSIENT
 $\Delta k = .0102$, $f_{as} = .002$

Figure 59.

POWER vs TIME SPERT STEP TRANSIENT $\Delta k = .0102$, $f_{as} = .005$

Figure 60.

SPERT STEP TRANSIENT
MAXIMUM FUEL TEMPERATURE vs TIME
 $\Delta k = .0087$, $f_{as} = .005$

Figure 61.

SPERT STEP TRANSIENT
MAXIMUM FUEL TEMPERATURE vs TIME
 $\Delta k = .0102$, $f_{as} = .002$

Figure 62.

SPERT STEP TRANSIENT
MAXIMUM FUEL TEMPERATURE vs TIME
 $\Delta k = .0102$, $f_{as} = .005$

Figure 63.

HFBR TRANSIENT BEHAVIOR WHEN $\Delta k = .0100$,
 $P_0 = 10$ KILOWATTS, $T_0 = 50^\circ\text{C}$, $f_{as} = .002$

Figure 64.

HFBR TRANSIENT BEHAVIOR WHEN $\Delta k = .0125$,
 $P_0 = 10$ KILOWATTS, $T_0 = 50^\circ\text{C}$, $\text{fas} = .002$

Figure 65.

HFBR TRANSIENT BEHAVIOR WHEN $\Delta k = .0137$,
 $P_0 = 10$ KILOWATTS $T_0 = 50^\circ\text{C}$, $\text{fas} = .002$

Figure 66.

HFBR TRANSIENT BEHAVIOR WHEN $\Delta k = .0156$,
 $P_0 = 10$ KILOWATTS, $T_0 = 50^\circ\text{C}$, $f_{as} = .002$

Figure 67.

HFBR TRANSIENT ANALYSIS
 STEP REACTIVITY INPUTS
 PEAK POWER vs STEP HEIGHT
 WHEN $P_0 = 10$ KILOWATTS

Figure 68.

HFBR TRANSIENT ANALYSIS
 STEP REACTIVITY INPUTS
 MAXIMUM FUEL TEMPERATURE vs STEP HEIGHT
 WHEN $P_0 = 10$ KILOWATTS

Figure 69.

HFBR TRANSIENT BEHAVIOR WHEN $\Delta k = .010$
 $P_0 = 40$ MEGAWATTS, $T_0 = 50^\circ\text{C}$, $f_{as} = .002$

Figure 70.

HFBR TRANSIENT BEHAVIOR WHEN $\Delta k = .0125$,
 $P_0 = 40$ MEGAWATTS, $T_0 = 50^\circ\text{C}$, $f_{as} = .002$

Figure 71.

HFBR TRANSIENT BEHAVIOR WHEN $\Delta k = .0137$,
 $P_0 = 40$ MEGAWATTS, $T_0 = 50^\circ\text{C}$, $f_{as} = .002$

Figure 72.

HFBR TRANSIENT BEHAVIOR WHEN $\Delta k = .0156$,
 $P_0 = 40$ MEGAWATTS, $T_0 = 50^\circ\text{C}$, $f_{as} = .002$

Figure 73.

40 MW
HFBR TRANSIENT ANALYSIS
STEP REACTIVITY INPUTS
PEAK POWER vs STEP HEIGHT

Figure 74.

40 MW
 HFBR TRANSIENT ANALYSIS STEP REACTIVITY
 INPUTS: MAXIMUM FUEL TEMP. vs STEP HEIGHT

Figure 75.

HFBR TRANSIENT BEHAVIOR (ANALOG RESULTS)
 STEP INPUT 50¢, $P_0 = 40$ MEGAWATTS
 $T_0 = 50^\circ\text{C}$, $f_{as} = .002$

Figure 76.

HFBR TRANSIENT BEHAVIOR (ANALOG RESULTS)

STEP INPUT 1\$, $P_0 = 40$ MEGAWATTS

$T_0 = 50^\circ\text{C}$, $\text{fas} = .002$

Figure 77.

HFBR TRANSIENT BEHAVIOR (ANALOG RESULTS)

STEP INPUT 1.50 \$, $P_0 = 40$ MEGAWATTS

$T_0 = 50^\circ\text{C}$, $\text{fas} = .002$

Figure 78.

HFBR TRANSIENT BEHAVIOR
RAMP INPUT 10 ϕ PER SECOND
 $P_0 = 1$ MEGAWATT, $T_0 = 50^\circ\text{C}$, $f_{as} = .005$

Figure 79.

Figure 80. Control rod reactivity rate graphs.
The rod speed is 4 in./minute.

Figure 81. Startup accident from 1 kw. A continuous ramp input of .10 \$/sec is imposed on the clean, cold reactor at 1 kw initial power. The shutdown is from a 60 Mw high level trip, with a 30 msec clutch delay time, followed by the drop of the safety rods from full-out (-5% k in 160 msec).

Figure 82. Startup accident from .01 watt. A continuous ramp input of .10 \$/sec is imposed on the clean, cold reactor at .01 watt initial power. The shutdown is from a 60 Mw high level trip, with a 30 msec clutch delay time, followed by the drop of the safety rods from full-out (-5% k in 160 msec).

Figure 83. Central thimble flooding accident. A step input of 0.5% k is imposed on the reactor at 40 Mw initial power. The shutdown is from a 60 Mw high level trip, with a 30 msec clutch delay time, followed by the drop of 3 safety rods from full-out (-2% k in 160 msec).

Figure 84. Cold water accident with no safety rod action. The reactivity input is a step of $.00245 \Delta k$ and a ramp of $.00176 \Delta k/\text{sec}$ ($.23 \text{ } \$/\text{sec}$). The initial power and temperature are 1 Mw and 56°C , and $f_{as} = .002$.

14 ELEMENTS
 $\Delta K \sim -10 \phi$

14 ELEMENTS
 $\Delta K \sim \text{SUBCRITICAL}$

14 ELEMENTS
 $\Delta K \sim \text{VERY SUBCRITICAL}$

LOADING ARRAYS (a), (b), & (c) ALL HAVE 14 CLEAN, COLD ELEMENTS OF 260 GM U^{235} EACH. IN EACH CASE THE ROD PATTERN HAS 2 MAIN RODS & THE MATCHING 2 AUXILIARY RODS FULL-IN, AT THE POSITIONS SHOWN.

17 ELEMENTS
 $\Delta K = 0$ WITH
 ROD 2 AT 21.1 IN.

18 ELEMENTS
 $\Delta K = 0$ WITH
 ROD 2 AT 12.7 IN.

19 ELEMENTS
 $\Delta K = 0$ WITH
 ROD 2 AT 12.1 IN.

LOADING ARRAYS (d), (e), & (f) HAVE 17, 18, & 19 CLEAN, COLD ELEMENTS OF 260 GM U^{235} EACH. IN EACH CASE THE ROD PATTERN HAS ROD 1 = MAIN ROD FULL-IN (0 IN.); ROD 2 = MAIN ROD AT POSITION INDICATED; RODS 3 & 4 = MAIN + AUXILIARY RODS FULL-IN (0 IN. & 0 IN.).

Figure 85. Hold-down experiments at various reduced loadings with rods spaced 1/2 in. outside the "J" core shroud.

Figure 86. Loss of pressure accident. At time zero the reactor system pressure is reduced to atmospheric. Initial conditions are 40 Mw power and 50°C core inlet temperature. The steam fraction, f_{as} , is .002.

Figure 87. Fuel handling accident total dosage as a function of distance from the reactor stack, calculated from Sutton's equation with $\bar{u} = 3.0$ m/sec, $C_y = .15$, $C_z = .40$, $n = .25$, and $h = 108$ m.

Figure 88. Integrated cloud centerline thyroid dose.

Figure 89. Direct integrated dose.