FINAL SAFETY ANALYSIS REPORT ON THE BROOKHAVEN HIGH FLUX BEAM RESEARCH REACTOR

J.M. HENDRIE

April 1964

OFFICIAL USE ONLY UNTIL DECEMBER 1, 1964

**BROOKHAVEN NATIONAL LABORATORY
UPTON, NEW YORK**

SECTION 8. EXPERIMENTAL FACILITIES

8.1 DESCRIPTION OF FACILITIES

8.1.1 BEAM HOLES. The main experimental facilities in the HFBR are the horizontal beam holes for external neutron beam experiments. There are nine such beams, with the characteristics of the various holes adjusted to suit the particular types of experiments to be done at each hole. The arrangement of the beam tubes in the reactor vessel is shown in Figure 8.1. This figure also contains a table listing all of the experimental facilities, their locations in the vessel, and the internal dimensions in the high neutron flux region.

The typical arrangement of the beam facilities in the biological shield is shown in Figure 8.2. Detailed plan views of several of the beam tubes are shown in Figures 8.3 and 8.4. Typical construction details for the beam tube and shutter case assembly are shown in Figure 8.5. Construction details of the beam plug are shown in Figure 8.6 and of the shutter in Figure 8.7.

Of the nine beam facilities, five are for thermal neutron beams. These are the facilities numbered H-3, H-4, H-5, H-6, and H-7. These tubes are arranged so that they do not "look" at the reactor core, and are sufficiently distant from the edge of the core so that the fast neutron background is greatly reduced and the predominant neutron energy in the beams is that of the thermal group (8.1). Two of the thermal beams, H-4 and H-6, are provided with large diameter liner sections in the biological shield to permit two neutron beams to be extracted from a single thimble in the reactor vessel. These facilities are called "dual thermal beams" in the table of Figure 8.1.

The beam facilities H-1 and H-8 are tangential beam tubes of standard dimensions, placed nearer the reactor core than the thermal tubes to increase the epithermal component of the emerging neutron beam. The facility H-1 is located quite close to a corner of the core and will have the harder energy spectrum of the two. The facility H-2 is a radial beam tube which extends to within a few inches of the edge of the reactor core. It will be used in conjunction with a long flight path fast chopper apparatus.

The facility H-9 is a large diameter thimble in the reactor vessel, 12 in. I.D., which may be provided with a refrigerated moderator for the production of an enhanced cold neutron beam. The initial experiments to be done with the H-9 facility do not entail a refrigerated moderator, and simply use a neutron filter to obtain the existing low energy neutron component from the reflector of the reactor.

The portion of each beam tube facility which is cast into the biological shield consists of a port box, stepped tube, and shutter liner. The latter two components were factory assembled to insure alignment and to minimize field welding. The port boxes are inset approximately 1 ft from the shield face and vary in size from 23 in. square for H-1, H-3, H-5, H-7, and H-8, 35 in. square for H-4, H-6, and H-9, to 72 in. square for H-2. These boxes serve several purposes; their use to enhance the shielding of the experiments is described in Section 8.3, and their function as a terminal point for H₂O plug cooling and D₂O experimental sample cooling lines is noted in 8.2. There are a pair of 3 in. pipes in the top and bottom of the port box which connect to the upper and lower chases, respectively. Through these pipes can be run power, gas, or vacuum lines from sources anywhere within the building.

The stepped tubes are accurately machined to a surface finish of 32 RMS for the larger diameter and 63 RMS for the small diameter. The tolerance and concentricity of the bores are carefully controlled to maintain a .090 in. diametral clearance with the plug. The same careful attention is paid to the fit between the lower portion of the shutter and its casing. Here a 9/64 in. maximum clearance is maintained all around. The stepped tubes and shutter casing are fabricated from type 304 stainless steel plate. The containment flange, which does not come in contact with the plug, is a forged carbon steel ring.

There are drain fittings in the port box and shutter liner which connect to a manifold and thence to liquid-gas separators. If the effluent is water from a leaky shutter or shield plug, it will drain to the GA-304 "D" waste pump. Gas flow through these lines is vented through the separators to the building exhaust system.

8.1.2 SHUTTERS. Each of the nine beam facilities is equipped with a shutter which can be moved vertically to close off the beam. A shutter consists of an upper heavy concrete shielding block and a lower water and steel filled box which is attached to the block. The location of the shutters and their dimensions are shown in Figures 8.3 and 8.4. An elevation of the shutter blocks in the biological shield is shown in Figure 8.2. Construction details of a typical shutter are shown in Figure 8.7. All of the shutters are 30 in. in length along the beam direction. The width varies from facility to facility, ranging from 7 in. for the standard beam holes to 20-1/4 in. for H-9. The shutter assemblies range in height from 6 ft-11 in. for beam tubes located above the core to 7 ft-11 in. for those below, and weigh from 3,000 lbs for H-2 and H-7 to 10,000 lbs for H-9.

The lower portion of the shutters, in the region of the neutron beam when the shutter is down, is composed of layers of 1% boron stainless steel in light water. The water is circulated past the steel plates in succession for cooling purposes. The steel plate assembly is enclosed in a water-tight type 304 stainless steel rectangular box the height of which varies from 11-1/2 in. for the smaller shutters to 19 in. for H-4, H-6, and H-9. Thermal neutrons are captured in a boron plate at the end of the water box facing the core to reduce the capture gamma ray burden on the shutter. The volume ratio of steel and water in the lower shutter is 60% steel and 40% water. The upper portion of the shutter block is filled with heavy concrete in a stainless steel box. Appropriate lifting lugs are provided at the top of the block for rigging.

Each shutter is raised and lowered by means of an electrical motor operator of the non-rising stem type. The stem is a lifting screw having an Acme thread which engages a bronze ball nut mounted upon the shutter. These drives units are located in wells just beneath the pit seal floor at elevation 120 ft-8 in. Gasketed cover plates in the seal floor permit access to the shutter motors for maintenance. Heavy duty operators for the three large shutters, H-4, H-6, and H-9, deliver 5 ft-lbs of torque at 15 rpm which is translated to a vertical shutter speed of 3.75 in./min. A torque of 3 ft-lbs is produced for the remaining, lighter shutters, rotating the lead screw at 13.8 rpm and moving the shutters at 3.45 in./min. Although the lighter shutters can be raised 12 in., H-5 and H-7 raised 19-1/2 in., and H-8, 21 in., the vertical motion will normally be no greater than the minimum required to clear the collimator openings in the stepped plugs.

Electric motors on the operators are 220 VAC, 3 phase, with class H radiation resistant windings and leads. Integral 2-position limit switches can be set to determine the length of shutter travel, and a torque overload switch

prevents motor damage should jamming or switch failure occur. The shutters are individually controlled from stations on the experimental floor of the building at the column line. Indicator lights at these control stations show whether the shutters are raised or lowered. The control station boxes are kept locked. The keys can be obtained by qualified personnel from the Shift Supervisor.

The shutter cooling water pipes extend upward through holes in a stationary shield plug mounted above the movable shutter block. The lifting screw also passes through a hole in this shield plug. The cooling pipes, which are of type 304 stainless steel, rise through packing glands in the seal floor. As the shutter moves up and down, the cooling water pipes slide up and down through the shaft seals. Several feet above the seal floor the pipes are connected to the cooling water manifolds through flexible stainless steel hoses. These are shown in Figure 8.2.

8.1.3 THERMAL IRRADIATION THIMBLES. There are three thermal irradiation thimble facilities in the HFBR, numbered V-10, V-11, and V-12. The location of these thimbles in the reflector portion of the reactor vessel is shown in Figure 8.1. Figure 8.8 is an elevation of the biological shield, showing the installation of a complete thermal thimble. Samples may be placed in these facilities for irradiation at high thermal fluxes for the production of high specific activity radioisotopes and similar uses.

The thermal irradiation thimbles are essentially straight pipes which are placed in the re-entrant tubes in the reactor vessel. The thimbles extend upward through the thermal shield cover, seal floor, and biological shield to sample charging stations at the 149 ft elevation. Each thimble assembly is made up of a lower flow tube and an upper charging tube. The lower tube is flange-connected to a remotely operated ball valve about 2 ft above the seal floor in the reactor pit service area. The upper charging tube extends from the valve to the pile top where it terminates in a 28 in. flange.

The lower tube is a 2-1/2 in. Schedule 40 6061-T6 aluminum pipe closed at the lower end with a welding cap. It is fitted with an internal tube of similar material, 2 in. O.D. x .065 in. wall, which serves as a flow divider. Heavy water enters a flanged tee located just below the ball valve and flows down the tube and through a grill plate at the bottom of the flow divider. The coolant returns up the .235 in. annulus between the tube and pipe to an outlet nozzle close to the inlet line.

The charging tube is a 2-1/2 in. Schedule 40 stainless steel pipe bolted to the ball valve at its lower end and seal-welded to a 28 in. 300 psi stainless steel flange at the top. The pipe is imbedded in a 7 ft-9 in. long stepped shielding plug filled with heavy concrete and welded to the underside of the flange. The otherwise straight thimble pipe is bent on a radius of about 7 ft to meet the flange squarely and thus simplify sample charging and discharging operations. A cask or blind cover flange is gasketed and bolted to the charging tube flange to seal the system. The ball valve can be closed to seal the thimble system, making it possible to operate the reactor or continue an irradiation experiment with the cask removed. The ball valve is 3 in. size, rated for 300 psi service and constructed of type 316 stainless steel. A 4-way solenoid pilot valve is used to control a pneumatic double-acting operator. Appropriate controls are located in cabinets at the pile top.

The thimble in the reactor vessel into which the irradiation tube assembly fits is similar to the horizontal beam tube thimbles, having an inside diameter of 3.5 in. The thermal irradiation tube slides into the vessel thimble. Since there is only intermittent line contact between the two thimbles, the heat

transfer from the irradiation tube assembly to the reactor vessel thimble is relatively poor. Thus, cooling water must be present in the tube assembly during reactor operating periods whether or not an irradiation experiment is in progress. A relatively low flow rate is sufficient to remove the heat generated in the walls of the tube.

Where the lower thimble tube passes through the seal floor, zippered neoprene boots are used to prevent CO₂ leakage into the upper pit. Clearance holes for the tubes through the thermal shield cover must, by necessity, be large enough for assembly purposes. Lead shield collars are used to reduce streaming up the outside of these tubes. Since the reactor vessel thimbles are essentially deep wells without drains, drip caps have been mounted on the tubes to prevent any condensate from entering the vessel thimble.

The coolant for the thermal irradiation thimbles is heavy water, supplied from the separate process system which is discussed in Section 8.2. The upper portions of the irradiation thimbles and the loading casks are normally filled with helium at atmospheric pressure.

8.1.4 FAST IRRADIATION THIMBLES. There are four thimbles in which fast neutron irradiations of small samples may be carried out. Two of these are located in the center of the core, V-15 and V-16, and two are located at the edge of the core, V-13 and V-14. These are shown in Figure 8.1. An elevation of the assembled fast irradiation thimbles in the reactor vessel and biological shield is shown in Figure 8.8. A fast irradiation thimble is composed of a lower tube assembly, a charging tube assembly, and an extension section in the top shield plug. The lower tube assembly and the charging tube are bolted together with a remotely operated ball valve between them.

The fast thimble terminations are part of the top plugs of the shield. A 28 in. 300 psi stainless steel flange identical to those used for the thermal facilities is cast into the appropriate top shield plug for each fast thimble. The extension tube is a 3 in. Schedule 40 stainless steel pipe welded to the 28 in. flange and terminating in a recess at the bottom of the shield plug. The extension tube is bolted to a stainless steel bellows, and the bellows is bolted to the charging tube. The differential thermal expansion between the reactor vessel and the top shield structure is absorbed in this bellows. The charging tube is a doubly bent 3 in. Schedule 40 stainless steel pipe which connects the lower tube assembly to the bellows. The "S" bend is made with large radii to provide maximum clearance for irradiation samples.

The lower tube is constructed with a double-walled section for the direction of cooling water downward in the central passage and upward in an annulus. The outer member of the assembly, the pressure tube, is a 2 in. Schedule 80 6061-T6 aluminum pipe closed at the bottom end with a weld cap of the same size and schedule. The inner tube is 1.5 in. O.D. by .065 in. wall and is also fabricated of 6061-T6 alloy. Inlet and outlet nozzles for the heavy water coolant are located at the upper end of the tube assembly. Just below the inlet and outlet nozzles is a heavy aluminum flange which is bolted to the top closure assembly of the reactor vessel. This flange seals the hole through which the re-entrant tube assembly is passed into the reactor vessel and makes a rigid connection of the fast thimble to the reactor vessel.

The 2 in. stainless steel ball valve serves the same functions as the 3 in. thermal facility valves. These valves are rated for 300 psi service and are remotely operated from switches at the pile top. The valve operators are double-acting pneumatic types, controlled by 4-way solenoid valves.

The in-core fast thimbles extend to a point 13-1/8 in. below the center-line of the core. In the region of the fuel elements they pass inside cylindrical flow shrouds which provide lateral support and locate the thimbles in the center of the core (see Section 4.5.1). These shrouds are fitted at the top ends with rectangular boxes similar to those of the fuel element upper ends. At the lower end each shroud has a fuel element-type lower box and fits into the grid plate. If the reactor is to be operated without a fast irradiation thimble in a central position, a substitute flow-blocking shroud must be substituted for the standard one.

The core-edge irradiation thimbles pass through the stainless steel transition plate above the core. A close-fitting collar arrangement is provided where these aluminum pipes pass through the plate. The collar arrangement reduces velocity of leakage flow which might otherwise erode the aluminum thimble (see Section 4.5.2). Again, if the reactor is to be operated with the core-edge thimbles removed, the apertures in the transition plate must be blocked to prevent bypass flow. This is done by a set of cover plates constructed for the purpose.

The fast thimbles are located in the reactor vessel and must withstand the primary system pressure. This is the reason for the relatively heavy-walled outer pipe on each of the thimble assemblies. Since the failure of one of the fast thimbles would allow the pressurization of the entire experimental heavy water system to primary system pressure, the experimental heavy water system is constructed to the same piping specifications as the primary, and is operated in such a way as to withstand the possible increase in pressure level from the normal few pounds operating pressure to full primary pressure.

Because the portions of the fast irradiation thimbles in the reactor vessel are immersed in the primary coolant, cooling requirements are minimal. An internal heavy water flow is provided for sample cooling. The internal flow divider tube is adequately cooled by the conduction of gamma heat through static heavy water to the outer tube wall and flow is not required in the thimble if it is not in use.

The quality control and test procedures used in the manufacture of the vertical experimental facilities were similar to those used for other components of the primary system. All pressure bearing pipes or tubes were ultrasonically tested by the fabricator. Every strength weld was subject to a liquid penetrant inspection as outlined in MIL-STD-271B (SHIPS). Welders were qualified to ASME Code requirements and samples of each type of weld were tested for strength, porosity, and inclusions. All strength welds in stainless steel were radiographically inspected and conformed to ASTM-E99-55T, Reference Radiographs for Steel Welds. The porosity of both aluminum and stainless steel welds was equal to or less than the standard "Fine-Degree B". The assembled pipes were subject to a hydrostatic test as outlined in Section VIII of the ASME Boiler Code, and a helium leak test as described in MIL-STD-271B (SHIPS).

8.2 EXPERIMENTAL FACILITY COOLING

8.2.1 INTRODUCTION. There are two cooling systems necessary for the operation of the experimental facilities. The first is a heavy water system which is used primarily for the seven vertical irradiation thimbles. The other cooling system contains light water and is part of the biological shield cooling system. It services the horizontal beam tube plugs and shutters. Auxiliary cooling apparatus is necessary for some experiments, and compressed or liquefied air, nitrogen, and helium will be used, as well as domestic cold water.

8.2.2 D₂O EXPERIMENTAL FACILITY COOLING. The heavy water experimental facility cooling system removes heat generated in the fast and thermal irradiation thimbles. It also makes available, adjacent to the beam tubes, heavy water for cooling experimental gear lying within the re-entrant beam tubes of the reactor vessel. Any cooling performed within the reactor cavity is done with heavy water.

For the first set of beam experiments planned for the HFBR, no internal radiators in the vessel beam tubes are contemplated. The neutron beams originate in the heavy water reflector at the ends of the various beam tubes. In the future it is possible that experiments may require radiator blocks in the beam tubes in regions of high radiation heating. Such radiator blocks will have to be cooled by heavy water supplied from the experimental heavy water system. The supply and return headers P-152 and P-139 are available in the upper chase of the biological shield for this purpose. The headers have not been carried completely around the upper chase since no need exists at the present time. Since the heating rate in a radiator block installed in a beam tube would be high compared to the heating rates in the shutters or plugs, appropriate flow and temperature instrumentation on the coolant stream for each radiator block will be necessary. This instrumentation will have to be arranged in two out of three logic systems and will be connected to the reactor setback or scram systems.

All seven of the irradiation thimbles are supplied with heavy water from the experimental facilities cooling system, shown in Figure 8.9. Individual supply and return connections for each thimble are made from the 4 in. headers P-143 and P-140 located in the process area of the operations level. Kates flow regulators, FCI-10 through FCI-16, along with temperature indicators TI-10 through TI-16 are installed in the return lines of each of the fast and thermal irradiation thimbles. Flow indicating alarms, FIA-10 through FIA-16, which will alarm on low flow are installed in the thimble supply lines.

The thermal irradiation thimbles receive 12 gpm each at a supply temperature of 120°F. The maximum heat rate in each thimble is estimated to be 45 KW. The two core-edge irradiation thimbles receive 17 gpm each at a temperature of 120°F. The maximum heat load in these facilities is expected to be 50 KW each. The two in-core thimbles receive 33 gpm each at a 120°F. The maximum heating rate in the in-core thimbles is about 100 KW each. The return water temperature is 140°F.

The pumps, heat exchanger and heavy water cleanup portions of the system are located in the process area on the operations level. Flow in the experimental facilities cooling system may be as much as 136 gpm, but will vary according to the number of experimental facilities utilizing heavy water cooling. Under normal operating conditions the experimental facility cooling system will operate at slightly above atmospheric pressure. However, the system is capable of withstanding primary system operating pressure, since a leak in the in-vessel thimbles would bring the primary system pressure into the experimental cooling system.

The experimental facility cooling system is equipped with two pumps, GA-103A and GA-103B, which are connected in parallel. Since one pump supplies the required flow, the second pump serves as a standby unit. Either pump may be used for continuous or standby service. Pertinent data on the pumps is given in Table 8.2-1.

The pump motors are supplied electricity at 440 volts, 3 phase, from a breaker in motor control center MCC-2. A disconnect switch for each motor is

Table 8.2-1 Data on Experimental Facilities Cooling System Pumps
GA-103A and GA-103B

Manufacturer	Pacific Pumps, Inc.
Type	Vertical, in-line, centrifugal
Rated Flow	175 gpm
Rated Head	71.5 ft
NPSH required at rated flow	4 ft
Number of Stages	1
Impeller	9-5/8 in. diameter enclosed type
RPM	1760
Motor	Louis Allis 7.5 hp, 440 volt, 3 phase, 60 cycle with special class "B" epoxy insulation
Seals per pump	1 (similar to GA-101A seal: see Section 5.3.2)
Code	API Standard 610
Materials of construction:	
Casing, impeller, inner case parts, shaft	Stainless steel, type 304, .06% max. carbon content
Wear parts	Stainless steel, type 304, as above, with Colmony #6 overlay

located near the pumps, with a green pilot light to show that the power is off. With the disconnect switches closed, the pumps are controlled from the reactor console by two 3-position switches. The 3-position switches are of the "hand-off-auto" type. The selector switch for the pump selected to run is placed on "hand". The standby pump switch is set on "auto". In this configuration the standby pump will start automatically should the first pump fail. Both pumps may be run simultaneously with the selector switches both on "hand". Red and green pilot lights on the console indicate the condition of each pump; red for running, green for off.

There is a single cooler in the experimental facilities cooling system, EA-102. It is a cooler of the U-tube type, mounted in a horizontal position. Table 8.2-2 gives data on this exchanger. A tank is mounted above the cooler to serve as a surge volume for the system. A cover gas of helium is supplied from the low pressure helium system and is controlled by hand valves in the process system area. A periodic purge of the cover gas will remove any contaminants in the helium. A relief valve set at 250 psig is connected to the surge tank.

The following is a list of tests to which the materials for the cooler, and the completed cooler, were subjected:

1. Hydrostatic testing with fluorescent dye was performed at 415 psig on the shell, surge tank, and tube bundle, and at 225 psig on the channel and tube bundle.

2. Eddy current testing was performed on the longitudinal welds of the tubes.
3. A 100% X-ray inspection was made of all shell-side welds.
4. Ultrasonic testing of all shell-side materials was performed.
5. A mass spectrometer leak test was employed to verify the leak-tightness of the shell side of the cooler.

Table 8.2-2 Data on the Experimental Facilities Cooler EA-102

Manufacturer	Southwestern Engineering Co.
Shell side fluid	D ₂ O
Tube side fluid	Cooling tower water
Rated flow:	
Shell side	175 gpm
Tube side	110 gpm
Operating temperatures:	
D ₂ O in	140°F
D ₂ O out	120°F
Cooling water in	85°F
Cooling water out	120°F
Design temperature	200°F
Design pressure:	
Shell side	250 psig
Tube side	150 psig
Operating pressure:	
Shell side (normal)	5 psig
Shell side (emergency)	250 psig
Tube side	45 psig
Pressure drop at rated flow:	
Shell side	5 psi
Tube side	8 psi
Heat transfer rate:	
Clean	400 Btu/hr-ft ² -°F
Service	220 Btu/hr-ft ² -°F
MTD (corrected):	21.7°F
Surface	403 ft ²
Heat exchanged at rated flow	1,923,000 Btu/hr
Materials of construction:	
Tubes 3/4 in. O.D. x 16 B.W.G.	ASTM A-249 stainless steel, type 304, .06% max. carbon, seam welded
Tube sheet, shell, surge tank	ASTM A-240 stainless steel, as above
Channel shell	ASTM-A212-B carbon steel
Channel cover	ASTM-A181-2 carbon steel
Number of active tubes	184 (92 "U" tubes)

Table 8.2-2 (cont'd)

Closure design:	
Shell to tube sheet	welded
Tube to tube sheet	rolled and seal welded
Channel to tube sheet	bolted and gasketed
and channel to channel cover	
Surge tank capacity	30 gallons
Code requirements	TEMA-R for entire cooler and ASME Unfired Pressure Vessel Code with Case 1270N for shell side only.

The experimental facility cooling system is provided with filters and ion exchange beds for the purpose of maintaining megohm quality of the coolant. The purification system is sized to remove particulate and ionic debris from the heavy water in the event of a large (several pound) irradiation sample failing and breaking up. Filters FD-103 and FD-104 are identical to the filters FD-101A and FD-101B described in Section 7.5, except that they have a 1/2 in. drain connection in the bottom of the filter vessel. The vessel for the mixed bed, H-103, is identical to the vessels for mixed beds H-102A and H-102B described in Section 7.5. The ion exchange medium used is a mixed nuclear grade resin containing equivalent quantities of cation and anion resins in the hydrogen and hydroxyl form. The experimental system heavy water is neutral, at pD 7.35.

All piping in the experimental facility cooling system is to the primary system standard, specifically classifications A3K or B3K as described in Section 5.3.4. The heavy water inventory in the experimental facility cooling system is 449 gallons. This figure includes the inventory in the irradiation tubes, but does not include heavy water which will be contained beyond the block valves at the beam tube manifolds. A detailed breakdown of the inventory by components is given in Table 5.2-1 in Section 5.2.2.

8.2.3 H₂O EXPERIMENTAL FACILITY COOLING. We shall be concerned here with the cooling of those portions of the nine horizontal beam facilities which lie outside the thermal shield. The re-entrant thimbles in the reactor vessel are cooled by the primary coolant in the vessel, as discussed in Section 5.5. The apertures in the thermal shield through which the neutron beams pass are lined with steel cylinders cooled by the thermal shield heat removal system (see Section 10.2). The elements of the beam facilities outside the thermal shield are the shutter case, the cylindrical beam tube liners, the shutters, and the stepped plugs.

The shutter cases and cylindrical tube liners are stationary members cast into the heavy concrete of the shield structure. Heat generated in these stainless steel members by radiation is conducted through the heavy concrete to the cooling coil array at the inner edge of the biological shield.

Both the shutter blocks and the stepped plugs are water cooled. Light water is supplied to the plugs and the shutters from the biological shield cooling system. This process system circulates demineralized light water through the shutters, beam plugs, and the cooling coils in the heavy concrete biological shield. The elements of this process system are discussed in Section 7.10.

Cooling water to the shutters is supplied from the 2 in. manifold TW-221 and is returned to the 2 in. manifold TW-224. Both of these headers are cast into the heavy concrete of the biological shield at about elevation 127 ft-6 in., as shown in Figure 10.1. Individual 1 in. supply and return connections from the headers to each shutter are brought into the upper pit in the shield. Flexible stainless steel hoses connect the stationary supply and return pipes to the moving water pipes on the shutter blocks. The total flow to the nine shutters is 100 gpm. The supply temperature is 94°F and the return temperature is only 1.5°F greater. The water is at essentially atmospheric pressure in the shutters.

Cooling water is supplied to the plugs in the beam ports from the 2 in. header TW-223, and is returned to the 2 in. header TW-225. These headers are located in the horizontal upper chase on the outer face of the biological shield. Individual supply and return connections are made to the headers above each beam tube port box. Tubing connections are made to the outer ends of the plugs. A typical cooling coil installation in a plug is shown in Figure 8.6. A total water flow of 75 gpm goes to the nine plugs. As for the shutters, the water supply temperature for the plugs is 94°F and the return temperature a degree or so higher. The water pressure in the plugs is a few lbs/in² above atmospheric.

The flow of cooling water to the shutters and beam plugs is necessary while the reactor is operating at power to prevent overheating of these components. Flow meters in the shutter and beam plug supply manifolds are arranged to sound an alarm in the control room should these flow rates fall below normal. In addition, the 2 biological shield coolant pumps, GA-203A and GA-203B, together with the shutter coolant flow meter, F1a-203, are arranged to provide a two out of three logic setback signal in the reactor safety system. The process system is also instrumented to give an alarm in the event of an excessively high return water temperature.

8.3 SHIELDING OF EXPERIMENTS

8.3.1 HORIZONTAL BEAM TUBES. For the horizontal beam tubes the shielding components of interest are the shutter, beam tube plug and collimator, supplementary shielding in the port box and service chases, and the external shields around experimental equipment. The lower portions of the shutters, which intercept the beam in the down position, are constructed of boron-loaded stainless steel immersed in light water with a view to maximum attenuation of fast neutrons. The lower shutter box is closely fitted to the shutter casing and machined flat on the bottom to reduce leakage. The 30 in.-long shutter, composed of about 60% by volume of steel and 40% of water, has a fast neutron attenuation factor of about 9.0×10^5 and a core gamma reduction factor of approximately 2.5×10^4 . With the shutter closed and the reactor operating at full power, radiation levels will be high at the port box. For the radial beam tube H-2 in particular, additional stopping material in the beam path will be necessary to reduce the external beam to a level at which equipment may be serviced.

The standard stepped shield plug consists of an outer, or annular plug of steel and heavy concrete with appropriate water cooling coils. It has a central stepped hole which will initially contain a solid plug of steel and heavy concrete. The combination of annular and solid plugs, together with the shutter in the closed position, gives a total attenuation in the beam tube which is very nearly that of the bulk concrete shield. The solid plug will eventually be replaced by a collimator assembly designed to suit a particular experiment. Figure 8.6 shows the standard annular plug for holes H-4, H-6, and H-9.

Many of the plugs for experiments differ from the design shown in Figure 8.6. In general, the experimental plug designs include some remotely controlled means of shutting off the beam in the plug itself. In some cases this is accomplished by rotating a barrel-type shutter within the plug and in other cases by flooding the collimator hole with light water. Without these additional shutters very brief access to external apparatus adjacent to the emerging beam is possible immediately after shutdown and unlimited access is possible 8 to 12 hours later with the main shutter closed. With the auxiliary shutter arrangement, complete access to the area in front of the beam hole should be possible soon after shutdown.

The supplementary shielding in the port box area is a function of individual experimental setups. Where possible, experimenters will stack shielding blocks of lead or concrete into those regions of the port box which are not otherwise occupied by apparatus. The principal function of the port box is to provide a recess in the shield face into which the external shielding array may be stepped, to reduce radiation leakage along the crevice at the face of the biological shield.

An arrangement of recessed horizontal chases and connecting vertical chases has been provided in the face of the biological shield to accommodate service lines. Light and heavy water manifolds run in the upper chase. It is possible that the N-16 activity generated in the biological shield cooling system water will give some trouble as a radiation source. If this is the case, shield blocks will be stacked around these pipes to reduce the background level.

Even with small apertures in the plugs for the emerging beams, and with small-angle collimation, the radiation level of the beams outside the biological shield is very high. The external experimental apparatus must, therefore, be heavily shielded. The requirements for each experiment are more or less unique, depending upon the beam aperture, the collimation, and any filtering which may be done in the beam. In general, external shield arrays amounting to 3 to 4 ft of heavy concrete are required around the experimental apparatus. As noted in Section 10, it is not feasible to attempt to calculate in advance all of the possible experimental geometries and to deduce standard values of radiation levels in the experimental area for all of the diverse experiments which may be done in the future. The control of the radiation level in the experimental area will be a matter of continuing survey work by the Health Physics Group.

8.3.2 VERTICAL IRRADIATION THIMBLES. Shielding of the irradiation thimble facilities is substantially simpler than for the beam experiments. All of the irradiation thimbles are amply shielded for the sections within the boundary of the biological shield. The shielding problem external to the biological shield then reduces itself to a matter of providing sufficiently heavy casks for the loading, unloading, and transportation of irradiated samples. The handling casks proposed for the initial operation of the reactor generally maintain a 12 in. thick lead shielding layer, or equivalent, around irradiated samples. This shielding thickness is sufficient to contain 10^4 curies of cobalt, and is well in excess of the expected shielding requirements. As new experiments are devised for the irradiation thimbles the shielding problems for active material will have to be examined in detail for each new experiment. Again it is impracticable to attempt to anticipate all of the kinds of experiments which may be done, and their individual shielding problems.

8.4 CONTAINMENT FEATURES

The containment of the horizontal and vertical experimental facilities includes several areas of interest. Both types of facilities are designed to withstand the pressure effects of any conceivable reactor accident. In normal operating conditions the plugs in the horizontal facilities are part of the reactor cavity gas seal. The vertical facilities contain heavy water and must be carefully vented to insure that heavy water vapor does not reach the building atmosphere.

In each of the horizontal beam tubes the primary containment element is the heavy annular plug. The plug is gasketed and bolted to the fixed flange at the port box face. The bolting is required to be strong enough to resist an internal pressure of 400 psig. With the plug bolted firmly in place on the sealing gaskets, a complete separation in each beam tube is established between the CO₂ atmosphere in the reactor cavity and the air atmosphere of the experimental area. The beam aperture through the plug is sealed by an aluminum cap through which the neutron beam passes. The seal cap is designed to withstand a cavity pressure of 400 psig. A central plug or collimator section inserted into the annular plug must also be retained by heavy bolting.

The intention of the heavy flanges and bolting, and of the high internal design pressure for the plugs, is to insure that no credible reactor accident will lead to plug components being projected from the beam tubes with sufficient force to cause damage to the building shell. The strength of the flanges and the bolting arrangements are sufficient to accomplish this objective.

There are additional circumstances which reinforce our conclusion that no missiles will arise from the beam tubes. During periods when a beam hole is not in use, the shutter block will be in the closed position. The shutter block will close off the beam tube, and greatly reduce any sudden loading of the plug structure. At such times as a beam tube is in use in an experiment with the shutter raised, massive external shielding is necessary immediately outside the shield face. This shielding block array would then stop any plug material expelled from the beam tube. The shutter itself, which might be subject to an upward force from a cavity overpressure, is restrained from rising out of its well by the overhead shield block. This block, in turn, is securely bolted to the steel structure of the shield.

The irradiation thimbles all terminate in heavy flanges in the same manner as the beam tubes. When the irradiation thimbles are in use, heavy loading casks are securely bolted to the end flanges and become containment members for the thimble internals. Thimbles not in use are blanked off at the top flange by a heavy steel plate. Since the irradiation thimbles all pass through the biological shield from the upper pit area, which has an air atmosphere, the thimble plugs are not required to be gas seal members. All of the components of the irradiation thimble system, including the loading casks and the experimental facilities process system are designed with the possibility of a failure of a fast thimble in the reactor vessel in mind. Such an internal failure of a fast thimble would cause the experimental facilities process system and all attached components to be pressurized to a primary system pressure. Thus, all of the casks and piping systems for the irradiation thimbles are designed for 300 psi service.

Each of the irradiation thimbles may be isolated by a remotely operated valve located just above the lower tube section of the thimble. The closing of these isolation valves separates the lower portions of the thimbles, which enter the reactor vessel or the thermal shield cavity, from the charging tube

and loading cask portions. On the process system side, valves in each cooling water supply and return line make possible the isolation of any given thimble from the others.

8.5 DESCRIPTION OF TYPICAL EXPERIMENTS

The horizontal beam tube experiments include a wide range of studies of matter in the solid state, vibrational spectra in solids and liquids, capture gamma ray spectroscopy, cross section measurements, and polarized neutron studies of heavy atom resonances. A typical experimental setup for a diffraction experiment is shown in Figure 8.11. Here the neutron beam is brought through a first collimator to impinge on a monochromating crystal. The first crystal is mounted in a cavity in a heavy shielding block. The scattered neutron beam at a particular energy then passes through a second collimator to the crystal of interest. A further scattering from the second crystal is then examined in angle and intensity with a counter mounted on a rotating arm. The primary beam is heavily attenuated in the massive shield of the first crystal and is finally completely absorbed in a backstop shielding cavity. A typical beam aperture for such an experiment might be .75 in. wide by 1.5 in. high. The first collimator angle might run from 5 to 30 minutes of arc.

Other experiments use rotating slotted wheels to chop the neutron beam into discreet segments. The time of flight of various neutrons from the chopper to a counter located some distance away is then a measure of the neutron energy. This "time of flight" technique is used in a variety of ways for energy analysis of scattered and transmitted neutron beams. Some of the current chopper experiments require three rotating slotted wheels all carefully phased together in a single experiment.

The chopper experiments may require counting stations well removed from the face of the biological shield. The experiment at H-2, in particular, is expected to eventually extend outside the reactor building. A 36 in. diameter flanged nozzle in the building wall has been provided for this purpose. The beam will pass through an aluminum seal plate at this nozzle. Like the blank steel plate it replaces, the aluminum window is designed to withstand a 2 psig internal pressure in the building, and will be gasketed and bolted to the nozzle flange to maintain the building containment.

Such experiments as are envisioned at the present time for the irradiation thimbles are of conventional character. These involve the insertion of small quantities of given materials in the high thermal flux field to obtain high specific activity radioisotopes. The initial fast radiation damage experiments will involve the irradiation of small single crystal specimens of various metals. Future experiments associated with the irradiation thimbles will undoubtedly grow more complex in terms of the instrumentation required in-pile and in terms of temperature control of the specimens.

An early modification to the irradiation thimble system which is contemplated is the installation of one or more hydraulic rabbit systems. In these devices the heavy water is used as both coolant and propellant, with the rabbit cylinder functioning as a loose-fitting piston. By reversing the flow direction, the sample can be discharged to the loading cask. The hydraulic rabbit systems will allow loading and unloading of sample capsules to the high flux region for those samples which do not require thermocouple leads or other instrument connections.

8.6 INSERTION AND REMOVAL OF EXPERIMENTS

8.6.1 HORIZONTAL BEAM EXPERIMENTS. At startup, each of the nine horizontal ports will be fitted with both an annular and a solid central plug. Soon after startup the solid plugs will be replaced with collimator plugs designed to suit specific experiments. In all cases where plugs are to be removed, the reactor must be shut down and the shutter closed. Replacement of the central plug in a standard annular plug does not involve breaking the gas seal to the reactor cavity. A small shielding cask is brought to the face of the pile and the old central plug is withdrawn and carried in the cask to a suitable storage area. The replacement plug is then loaded into the central hole. It is not felt that this procedure will be complicated since the central plug does not contain water cooling lines and does not weigh over 200 pounds.

The replacement of the larger annular plug by a new plug for a given experiment is a task requiring heavier equipment. For this replacement, the procedure once more starts by closing the shutter. Since the removal of an annular plug involves breaking the gas seal into the reactor cavity, some care must be taken to maintain proper ventilation in the beam tube. While the seal is broken a maximum exhaust draft is maintained on the reactor cavity. Further, the exhaust and drain lines from the bottom of the shutter box and from the port box are opened to draw air from the beam tube. The slightly lower gas pressure in the experimental cavity, together with these various exhaust lines drawing on the beam tube and reactor cavity, should insure a net inward movement of air and a minimum diffusion in the experimental area of radioactive gases from the cavity.

Prior to removal of the annular plug, the cooling water lines inside the port box are uncoupled and drained. Following this, the heavy cover flange is unbolted and withdrawn by means of a support frame mounted on a fork truck. A heavy duty table is put into place under the port box, bolted to the pile face, and the table carriage set to the appropriate height. The plug handling cask is then lowered upon the carriage and positioned accurately with respect to the stepped beam tube. A cylindrical tube within the cask, called the transfer cylinder, is driven across the port box until it engages the beam tube. Next, the annular plug, which rides upon 4 small wheels, can be pulled into the transfer cylinder, the cylinder retracted into the cask, and a vertically operated gate at the mouth of the cask closed.

With the old plug safely in the handling cask, the cask is moved (by the experimental area crane) to the plug storage vault. Here the old plug is inserted in a storage cavity in the heavy concrete vault. The new experimental plug is loaded into the beam tube from an appropriate cold plug handling tray. Once the plug is seated within the beam tube and the flange bolts tightened the gas seal is restored. Following such a change in annular plugs, with the attendant breaking of the gas seal, a slow purging of the reactor cavity with CO_2 is necessary to remove the bulk of the air which will have leaked in during the plug changing operation.

After the new plug is inserted in the beam tube and bolted home, the appropriate water connections and compressed air leads may be made up. Actuating motors and other equipment mounted in the port box may be installed, and the external shielding array and experimental apparatus moved into position.

For some experiments the stepped plugs may extend through the shutter box region. In these cases the shutter cannot be closed while the plug is in place. As such a plug is withdrawn into the handling cask, the shutter will be lowered as soon as the end of the plug passes out of the shutter case.

Similarly, in the reloading operation, the shutter will have to be raised as the plug is driven home in the beam tube.

Although none of the initial experiments contemplate insertion of radiator blocks into the vessel beam tubes, such arrangements are possible in the future. The plugs associated with these experiments will be considerably longer than the standard plugs and will contain coolant lines to the radiator. They will also be designed to permit partial closing of the shutter with the plug in place. The existing cask can accommodate a 5 ft-1 in. long plug and must be extended to almost double this length for the longer plug. The cask is designed with this expansion in mind. As in the case of the plugs which extend through the shutter box region, the shutter must be open during the insertion and removal of such an experiment. The loading must be done from the handling cask to obtain sufficient shielding with the shutter in the raised position.

8.6.2 IRRADIATION THIMBLE EXPERIMENTS. The handling of the irradiation experiments is of interest from two standpoints. The first of these is the matter of installing and removing the irradiation thimble components themselves. The other concerns the loading of irradiation samples into the thimbles.

The initial installation of the vertical thimbles is a straightforward assembly procedure. Care must be exercised in placing the lower tube sections to avoid damage to the thimbles or vessel. The removal of the irradiation thimbles is complicated by the fact that they are radioactive and may contain tritium-bearing vapor. The latter problem is resolved by the use of plastic suits and breathing apparatus for personnel disassembling the thimbles.

The removal of the irradiation thimbles involves draining and blowing out the coolant from the thimble proper. The thimble is isolated from the process system by valves in the operations area process room. The major sections of the irradiation thimbles, the charging tube assemblies and the lower tube assemblies, are decoupled by unbolting the flanges at the ball valve and the coolant inlet and outlet nozzles. Further unbolting of the 28 in. flange at the top of the biological shield then permits withdrawal of the charging tube assembly for the thermal irradiation thimbles. For the fast thimbles a further uncoupling is done to separate the charging tube from the bellows assembly. Handling of the charging tube assemblies will not be difficult since they will not be radioactive. The removal of the charging tube assemblies for the fast thimbles may be through the central fuel handling hole in the top plug or through the hole provided by the removal of any of the sector plugs.

The aluminum lower tube assemblies will be removed from the vessel proper or the thermal re-entrant tubes into a shield at the pile top. Those portions of the thermal thimbles located below the thermal shield, or of the fast thimbles in the core region, will be radioactive and must be handled with care. For the thermal thimbles, the lower tube assembly is simply withdrawn upward out of the thermal shield cavity after the seal at the upper pit floor is broken. For the fast thimbles, removal of this assembly requires unbolting from the vessel closure and the careful removal upward of the thimble assembly from the vessel.

The reactor may be operated with any or all of the irradiation thimbles removed. It is likely that the initial operation of the reactor will occur with a minimum number of fast thimbles installed.

The handling of samples into and out of the thimbles depends upon the nature of the specific experiment. Initial experiments will be made with

samples sealed in lightweight aluminum capsules. The capsule diameter is limited to 1-5/8 in. for the thermal thimbles V-10, V-11, and V-12, and to 1-1/8 in. for the fast thimbles V-13, V-14, V-15, and V-16. The insertion and removal of the capsule is done with a flexible tape controlled from the loading cask at the pile top. Cylindrical capsules of a maximum length of 12 in. can be handled in the thimbles.

Rabbit-type experiment handling equipment consists of a charging station bolted and sealed to the 28 in. flange at the pile top and a small portable cask resting upon the charging station. The charging station contains a valve lock to isolate the capsule from the coolant stream. A sample can be charged and discharged by controlling the direction of coolant flow in the thimble. Once the discharged sample has passed the valve lock, the lock is closed and the charging station is purged. The sample is then raised into the portable cask and the cask is removed to a hot cell.

Experiments involving capsules with instrument leads require a loading cask which contains a shielded reel for the leads, a shear to cut the leads from the discharged capsule, and a rotary basket to move capsules into position for unloading. The samples may either be removed from the loading cask to another cask for movement elsewhere, or the loading cask itself may be unbolted and moved. Helium cover gas is piped to the loading cask, and exhaust gases are vented to the building exhaust system.

8.7 REACTIVITY LIMITS AND SAFETY ASPECTS OF EXPERIMENTS

The most 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. The reactivity balance table in Section 4.6.5 lists a total allowance of 1.4 %k for control margin and experiments. Of this reactivity about 0.5 %k is reserved for the control margin. The remaining 0.9 %k is available for irradiation experiments. The total reactivity allowance in in-pile samples will be spread over several experiments so that a single sample should have a relatively small reactivity worth.

The possibility of rupturing one of the fast flux thimbles in the vessel has been recognized and the sample handling casks and all external process equipment and piping are designed to stand the full primary system pressure. Various malfunctions which might occur in the irradiation thimbles are listed in Table 8.7-1.

In view of the nature of the irradiation samples to be placed in the reactor, no nuclear coupling of the core and experiments will occur. This hazard exists in certain test reactor installations where full scale fuel element tests are performed in loops within the active core, but this type of installation is not used in the HFBR. The abrupt withdrawal of even a small sample, will, of course, give rise to a perturbation in reactor power. However, in view of the small reactivity worth associated with a single sample, the power perturbation to be expected even in a very abrupt sample withdrawal would not cause any damage to the reactor system, although it might lead to an automatic scram.

The remaining possibility for nuclear coupling between the experimental facilities and the core lies in a collapse of one of the beam tubes or thimbles. This event is considered in Section 14.2.2, where it is shown that the resulting transient would not lead to any core damage. The consequences of a

Table 8.7-1 Safety Aspects of Irradiation Thimbles

<u>TYPE OF ACCIDENT</u>	<u>HOW INDICATED</u>	<u>COUNTER ACTION</u>
A. In-Core and Core Edge Fast Thimbles (V-13, V-14, V-15, V-16)		
1. Thimble Rupture Within Vessel	1.a. High D ₂ O level alarm on EA-102 ² b. Pump GA-103A or B discharge pressure indicator shows rise	1. Shut down reactor. Replace defective tube, D ₂ O cleanup, if necessary
2. Thimble Rupture Outside of Vessel	2.a. Excessive flow on Flow Indicators, FIA-13 thru FIA-16 b. Leak detector sensing elements c. Low level on EA-102 d. Possible low pressure on pump GA-103A or B discharge indicators	2. Valve off Thimble. Discharge capsule, if possible. Schedule repair at shutdown
3. Loss of Flow	3.a. High temperature reading on TI-13 thru TI-16 b. No flow on flow indicators, FIA-13 thru FIA-16	3. Discharge capsule. Valve off thimble and replace at shutdown
4. Break in Capsule	4. Shown on conductivity recorder, CR-3, if sample is soluble	4. Valve off thimble. Discharge capsule, if possible. Decontaminate thimble or replace at shutdown
5. Capsule Jammed During Charging or Discharging	5. Sense of "feel" for attached capsules, differential pressure indicator on rabbit discharge station	5.a. If stuck in pig, remove pig to hot cell and dismantle b. If caught in "S"-tube, remotely dismantle and pull up into special shield c. If jammed in tube within vessel, pull tube as far as possible, cut and pull remainder into special shield or discharge to canal via the fuel chute
B. Thermal Irradiation Thimbles (V-10, V-11, V-12)		
1. Thimble Rupture within Re-entrant Tube	1.a. Excessive flow on flow Indicators FIA-10 thru FIA-12 b. Low D ₂ O level on EA-102	1. Shut down reactor. Discharge sample. Valve off thimble and replace defective tube
2. Items 2 to 5 similar to above	2. Indicated in same manner as items 2 to 5, above except that flow meters will be FIA-10 thru FIA-12 and the temperature indicators will be TI-10 thru TI-12.	2. This facility can be cut and discharged to the canal as in 5c, above.

collapse of one of the vessel beam tubes would be far more serious in terms of the curtailment of the research program while repairs were made than in terms of the immediate consequences of the minor nuclear transient which might occur.

References

- (8.1) H. J. C. Kouts, "Beam Tube Design for the High Flux Beam Reactor,"
J. Nuc. Energy, Parts A/B 17, 153 (1963).

Experimental Facility	Azimuth	Elev. From Center	Radial Dist. From Center	Inside Diameter of Tube
H-1 Fast Neutron Beam	040°	-6"	13"	3.5"
H-2 Fast Chopper Beam	088°	+6"	12.9"	3.5"
H-3 Thermal Beam	130°	-6"	18"	3.5"
H-4 Dual Thermal Beam	214°	+6"	18"	3.5"
H-5 Thermal Beam	245°	-6"	18"	3.5"
H-6 Dual Thermal Beam	280°	on center	18"	3.5"
H-7 Thermal Beam	315°	+6"	18"	3.5"
H-8 Intermediate Beam	353°	on center	16.8"	3.5"
H-9 Thermal or Cold Beam	158°	on center	24.3"	12"
V-10 Thermal Irradiation Thimble	046°	-7.1" (end)	28"	3.5" (vessel); 1.87" (sample tube)
V-11 Thermal Irradiation Thimble	090°	-7.1" (end)	29.4"	3.5" (vessel); 1.87" (sample tube)
V-12 Thermal Irradiation Thimble	323°	-8.1" (end)	23.5"	3.5" (vessel); 1.87" (sample tube)
V-13 Core-edge Irradiation Thimble	139°	-6.1" (end)	11.3"	1.37" (sample tube)
V-14 Core-edge Irradiation Thimble	259°	-6.1" (end)	11.3"	1.37" (sample tube)
V-15 In-core Irradiation Thimble	319°	-13.1" (end)	1.6"	1.37" (sample tube)
V-16 In-core Irradiation Thimble	349°	-13.1" (end)	1.6"	1.37" (sample tube)

Figure 8.1 Diagram of the beam tube and irradiation thimble layout in the reactor vessel.

Figure 8.2 Elevation of the biological shield assembly showing the experimental facilities.

PLAN OF HFBR BEAM TUBE H-3
TYPICAL OF THERMAL TUBES

PLAN OF HFBR BEAM TUBE H-4
TYPICAL OF DUAL THERMAL TUBES

Figure 8.3 Plan views of beam tubes H-3 and H-4, from shield face to core.

Figure 8.4 Plan views of beam tubes H-2 and H-9, from shield face to core.

Figure 8.5 Beam tube and shutter case assembly for H-4 and H-6. The general construction is typical of all beam facilities.

Figure 8.7 Detailed drawing of the shutter for H-6. The general construction is typical for all beam facilities.

Figure 8.8 Thermal and fast irradiation thimble assemblies in the shield and vessel.

Figure 8.9 Piping and instrument diagram for the experimental D₂O process system.

Figure 8.10 Typical diffraction experiment set-up at a thermal beam tube.

SECTION 9. INSTRUMENTATION AND CONTROL

9.1 SUMMARY OF REACTOR CONTROL SCHEME

9.1.1 INTRODUCTION. The two primary functions of the instrumentation and control system are to maintain the neutron flux level at the experimental facilities as nearly constant as possible during operating periods, and to protect the reactor against faults and malfunctions at all times. The various instruments and devices provided to carry out these functions are described in detail in Sections 9.2 through 9.7. This introductory Section, 9.1, summarizes the control scheme for the reactor and lists the various instrument systems and their functions.

9.1.2 REACTIVITY CONTROL. The reactivity of the system is controlled by two sets of neutron-absorbing rods. The maximum excess reactivity of the system is 10.6 %k during a normal power cycle, and is 13.4 %k for a completely fresh, cold core. The control rod reactivity effect is 37.5 %k, so that ample shutdown margin exists in the rods at all times. The temperature and void coefficients of the reactor are negative (see Section 4.6).

There are 16 control rods, divided into two banks or "crowns" of 8 rods each. The safety rods (top crown) are withdrawn to positions above the core to increase reactivity. In the fully-inserted position the safety rods blanket the sides of the core. The auxiliary rods (bottom crown) are withdrawn to positions below the core to increase reactivity. In the fully-inserted position the auxiliary rods overlap the lower end of the core by about 2 in. Both sets of control rods are placed outside the core, close to the position of maximum reactivity effect for thermal neutron absorbers. In the operation of the reactor the two sets of control rods are positioned at equal distances above and below the core center, thus achieving the required reactivity control with a minimum perturbation of the neutron flux in the experimental facilities. A description of the control rod drive mechanisms is given in Section 9.7, and the absorber blades are described in Section 4.3.

There are five separate modes of control rod operation, as follows:

1. Individual movement of any rod.
2. One auxiliary control rod may be controlled automatically.
3. Gang or crown movement of the eight safety rods or of the eight auxiliary rods. Both gangs may not be withdrawn simultaneously.
4. Setback action, which is the simultaneous insertion of all rods at the normal drive speed.
5. Scram action, which is the rapid insertion of the eight safety rods by stored mechanical energy, coupled with insertion of the auxiliary rods as in setback.

A supplementary system for reactivity control is provided for any situation in which large amounts of light water (H_2O) might enter the reactor vessel. This is the poison water injection system, described in Section 7.3 (see also Sections 4.6.12 and 14.2.4).

9.1.3 CONTROL STATIONS. The reactor is operated from a main control room on the operations level of the building. All important process and nuclear information is indicated or recorded in the control room, and all important valve and pump controls are located there. Instruments which do not indicate in the control room are limited to local pressure and temperature points which supplement the main information channels. Process elements which are not operated from the control room include pumps and valves in the canal water circulation system and the D₂O transfer system, valves in the D₂O purification system, and the secondary system pumps. Such building utility systems as air conditioning, heating, and lighting are also not operated from the control room.

9.1.4 INSTRUMENTATION. The instrumentation may be divided into 4 classes: nuclear, process or "non-nuclear", radiation monitoring, and safety system. The nuclear instrumentation consists of 8 channels which respond to the neutron flux level at the inner surface of the shield. Three channels derive power and period information from ion chambers for use in the safety system. Two channels indicate reactor power; one from ion chambers and one from a thermopile. One channel furnishes a signal from ion chambers for automatic power level control. The last two channels are counting rate systems for cold start-up, with fission chambers as the signal sources.

The non-nuclear instrumentation provides temperature, pressure, liquid level, and flow rate information from the various process systems. Appropriate process variable information from the primary, shutdown, thermal shield, and biological shield cooling systems is fed to the safety system to protect the reactor against low primary coolant pressure, flow rate, or liquid level, high or low core pressure drop, high primary coolant temperature, and loss of flow in the shutdown, thermal shield, or biological shield systems. Other process variable information from these and other systems in the plant is indicated or recorded in the control room, and alarm signals are given when operating limits are exceeded. Numerous local pressure and temperature indicators supplement the main process information channels on all systems.

The radiation monitoring instrumentation is composed of several subsystems. These are the area radiation monitors, the secondary water monitors, the primary fission product monitors, the stack monitor, the nuclear incident alarms, and the air monitoring systems. The area radiation and air monitoring systems are for personnel protection. The other systems detect off-normal contamination conditions in process systems or the exhaust air system, and give alarms so that suitable corrective actions can be taken.

The safety system is the circuitry which causes appropriate alarms or control rod actions in the event of off-normal nuclear, process, or radiation conditions. Information is generally supplied from 3 parallel channels for those variables which may cause control rod action, and the safety system includes 2 out of 3 logic circuitry for rod action.

9.1.5 RESPONSE TIME. The reactor imposes no special requirements on the instrument and control rod response times. The prompt neutron lifetime and delayed neutron fractions are typical of thermal D₂O reactors fueled with fully-enriched uranium (see Section 4.6.11). Conventional reactor instrumentation is, therefore, entirely adequate for the HFBR. The nuclear channels respond to a rise in reactor power at the trip level in about one millisecond. The process instrument response is slower, depending on the type of detecting element and the variable measured. The safety system has a 5 millisecond response time from the opening of the scram relay coil circuits to the breaking of the control rod drive clutch circuits. The safety rod mechanisms are capable of moving the rods from the full-out position to the top of the core (13 in. motion) in 175

milliseconds from the time of breaking of the clutch circuits. The reactivity effect which results from this motion is about 4.4 %k.

9.1.6 START-UP SOURCES. For the startup of a clean cold core a neutron source will be installed in one of the core facility tubes to provide an adequate neutron level. The source is removed from the reactor after startup. For a normal startup, in which one-half the core (14 of the elements) contains fission products, the photoneutron source strength will be sufficient to give "on-scale" readings on all neutron sensing devices (except the thermopile) and an auxiliary source will not be necessary.

9.2 OPERATING CONTROL STATIONS

9.2.1 CONTROL ROOM. The control room is located on the top floor of a two-story structure on the operations level. The arrangement and location of the control room are shown on the building drawings, Figures 3.2 and 3.6. The instrument shop and an equipment room are located beneath the control room. A window in the front wall of the control room looks out over the operations floor and the top of the reactor shield. Access to the control room will be limited to authorized personnel.

Remote instrumentation and system controls are utilized throughout the plant to permit complete control of the reactor and the essential associated systems from the control room. Instrument panels display the process and nuclear variables required for proper plant operation. When seated at the control desk, the operator has an unobstructed view of the instrument panels, and by means of the controls located on the console desk can fully control the reactor and associated systems during all phases of reactor operation. The controls on the console permit the operator to control the primary system pressure, operate all primary and support system pumps, establish and maintain the desired primary and secondary flows, operate the control rods (including the automatic mode), and cause immediate shutdown of the reactor in the event such action is required.

A detailed list of the various instrument displays, reactor controls, and equipment panels is given in the 11 subsections which follow. The arrangement of these units in the control room is shown in Figure 9.1.

9.2.1.1 Control Console. The control console desk arrangement is shown in Figure 9.2. The console is a three-section desk, each section having a horizontal and an inclined surface. The middle horizontal section is a writing surface. The other surfaces support instruments and controls. Reactor power level, differential power, primary coolant flow rate, and the start-up counting rate are indicated on the console by meters. The following control elements are mounted on the console, within easy reach of the operator:

- a. Manual scram button and lockdown switch.
- b. Main and auxiliary rod selector switches and control switches.
- c. Primary system pump start and stop buttons, with indicating lights.
- d. Shutdown system pump "on-off-automatic" switches, with indicating lights.
- e. Thermal shield system pump "on-off-automatic" switches, with indicating lights.
- f. Biological shield system pump "on-off-automatic" switches, with indicating lights.
- g. Experimental facilities cooling system pump "on-off-automatic" switches, with indicating lights.
- h. Primary system throttle valve open and close controls, with position

- indicators and limit lights.
- i. Primary pump suction valve open and close limit lights.
- j. Secondary system throttle valve open and close controls, with limit lights.
- k. Primary system pressure controllers.
- l. Amplifier range switch for nuclear power channel #4.
- m. Automatic power regulating channel controllers, amplifier, range switch, manual-automatic selector switch, reset control, and alarm lights.
- n. Fission chamber positioner control switches, with position indicators.
- o. Annunciator "acknowledge", reset, and test switches.

A building paging microphone, a house telephone handset, a headset jack, and a laboratory system telephone are also located on the console.

9.2.1.2 Instrument Panels. There are 13 instrument panels and 2 wedge-shaped spacer units which make up the main instrument display in front of the console. These panels are numbered as shown in Figure 9.1. Panels 1-5 contain non-nuclear instruments, panel 6 contains the control rod position indicator, panels 7-10 contain nuclear instruments, and panels 11-13 contain radiation monitoring instruments. The list of individual instruments on each panel is given below.

Panel 1:

- a. 24-point temperature indicator, monitoring 10 active points with 14 as spares (TI and TIa 10-19).
- b. PI-114, cover gas pressure in D₂O storage tank FA-101.
- c. PI-115, cover gas pressure in D₂O tank FA-102.
- d. LI-104, liquid level in D₂O storage tank FA-101.
- e. LI-105, liquid level in D₂O storage tank FA-102.
- f. LIa-103, liquid level in experimental facilities cooling system heat exchanger EA-102.
- g. FI-103, flow rate through D₂O cleanup beds.
- h. GA-101A and GA-101B pump motor ammeters.

Panel 2:

- a. 12-point temperature recorder, 11 active points and 1 spare (TR and TRA 1-11).
- b. 8-point conductivity recorder, 6 active points and 2 spares (CR and CRA 1-6).

Panel 3:

- a. FRa-300, secondary flow rate to main heat exchangers.
- b. FRa-101, flow rate in shutdown cooling system.
- c. FIa-301, secondary coolant flow at inlet to cooler EA-103.
- d. PIa-101a, primary coolant pressure at outlet from cooler EA-101A.
- e. PI-301a, secondary coolant pressure at inlet to cooler EA-101A.
- f. PIa-101b, primary coolant pressure at outlet from cooler EA-101B.
- g. PI-301b, secondary coolant pressure at inlet to cooler EA-101B.
- h. PIa-201, manifold pressure in low pressure helium system.
- i. PIa-202, manifold pressure in high pressure helium system.
- j. PI-102A, primary coolant pressure at discharge from shutdown pump GA-102A.
- k. PI-102B, primary coolant pressure at discharge from shutdown pump GA-102B.
- l. PI-302, secondary coolant pressure in main supply line from cooling tower.
- m. PI-303, secondary coolant pressure in main return line to cooling

tower.

- n. LI-203, liquid level in biological shield cooling system surge tank FA-203.

Panel 4:

- a. FRa-201, lower thermal shield coolant flow rate.
- b. FI-202, upper thermal shield coolant flow rate.
- c. FI-207, thermal shield coil coolant flow rate.
- d. 2-point temperature recorder, TRA-101 and TR-102, primary coolant temperatures at reactor inlet and outlet, respectively.
- e. TIA-103, primary coolant temperature at outlet from cooler EA-101A.
- f. TIA-104, primary coolant temperature at outlet from cooler EA-101B.
- g. FIA-203, coolant flow rate to shutters.
- h. FIA-204, coolant flow rate to shield plugs.
- i. FIA-205, coolant flow rate to concrete cooling coils.
- j. LIA-201, liquid level in thermal shield expansion tank FA-201.

Panel 5:

- a. Flow and power recorders FRa-100 and XR-100, primary coolant flow rate and heat balance power, respectively.
- b. PRCa-101, reactor vessel cover gas pressure.
- c. PdRa-101, primary coolant differential pressure across reactor core.
- d. LRA-101, liquid level in reactor vessel.
- e. LIA-106, liquid level in reactor vessel.

Panel 6:

- a. Control rod position indicators for 8 safety rods, with limit lights.
- b. Control rod position indicators for 8 auxiliary rods, with limit lights.

Panel 7:

- a. Power recorder, nuclear channel 5, thermopile.
- b. Power recorder, nuclear channel 6, automatic power level control.
- c. Linear current amplifier, channel 6.
- d. Linear current amplifier, channel 4, power indicator.

Panel 8:

- a. Power recorder, safety system channels 1, 2 and 3.
- b. Period recorder, safety system channels 1, 2 and 3.
- c. Log count rate meter, channel 8, fission chamber.
- d. Linear pulse amplifier, channel 8.

Panel 9:

- a. Safety amplifier, channel 1.
- b. Safety amplifier, channel 2.
- c. Safety amplifier, channel 3.
- d. Safety monitor.

Panel 10:

- a. Period amplifier, channel 1.
- b. Period amplifier, channel 2.
- c. Period amplifier, channel 3.
- d. Log count rate meter, channel 7, fission chamber.
- e. Linear pulse amplifier, channel 7.

Panel 11:

- a. Log count rate meter for RRa-100, fission product water monitor, primary coolant system.

- b. Log count rate meter for RRa-105, fission product gas monitor, helium cover gas in reactor vessel.
- c. 2-point recorder for RRa-100 and RRa-105.
- d. Log count rate meter for RRa-301A, gamma activity in secondary water from primary cooler EA-101A.
- e. Log count rate meter for RRa-301B, gamma activity in secondary water from primary cooler EA-101B.
- f. 2-point recorder for RRa-301A and B.
- g. Log count rate meter for RRa-302, gamma activity in secondary water from shutdown system cooler EA-103.
- h. Log count rate meter for RRa-303, gamma activity in secondary water from experimental facilities system cooler EA-102.
- i. Log count rate meter, spare.
- j. 3-point recorder for RRa-302, RRa-303, and spare.
- k. 2-point recorder for RRa-305 and RRa-306, stack gas and particulate monitors.

Panel 12:

- a. Ten area radiation monitoring system meters, for 10 stations located throughout the building.
- b. Check source panel.
- c. Power supply panel.
- d. Two nuclear incident alarm meters, 2 stations on balcony level in experimental area.

Panel 13:

- a. Pump start and stop buttons, GA-303, cooling tower emergency return pump, with indicating lights.

9.2.1.3 Annunciator Panels. Visual and audible alarms are provided in the control room for all variables which may cause control rod action and for all variables which require the prompt attention of the operator in off-normal conditions. The visual alarms are given by a set of 3 in. x 3 in. lighted windows. These elements are arranged in three annunciator panels which are mounted on top of the instrument panels (see Figure 9.1). The annunciator panel above instrument panels 1-3 contains the alarm points for variables which do not cause control rod action automatically. Figure 9.3 shows the face display for this panel, with the legends which appear for each variable. The scram and setback annunciator panel face displays are shown in Figure 9.4. The scram annunciator panel is located above instrument panels 7 and 8, and the setback annunciator panel is above instrument panels 9 and 10. The arrangement of the scram and setback panels is such that instruments which are part of each 2 out of 3 logic circuit are grouped in vertical columns. There are a total of 96 annunciator points, 27 in the scram panel, 21 in the setback panel, and 48 in the alarm panel.

The audible alarms associated with the annunciators are a horn for the scram and setback panels and a bell for the alarm panel. The sequence of operation of an annunciator point is as follows. When the instrument channel controlling the annunciator point signals an off-normal condition, the horn or bell sounds and the annunciator window is lighted by a bright, flashing light. The Operator may turn off the audible alarm by pressing an "acknowledge" switch on the console. The window light stops flashing when he does so, remaining bright and steady. When the off-normal condition is cleared, the annunciator light changes to a flashing mode, but at a reduced intensity, notifying the Operator that the point may be reset to normal. The Operator can then press the "reset" switch on the console and turn off the dim, flashing light, leaving the annunciator point in the normal, or "off" condition.

The annunciators are "De-Line Magna-plac, AR" units manufactured by the Scam Instrument Co. They are equipped with "lock-in" circuitry so that the alarm condition is held by the annunciators even if the off-normal signal clears itself before the operator can act. A test switch for the annunciator panels is located on the console with the acknowledge and reset switches.

Additional alarms are located in the radiation monitoring equipment and in the console. The area radiation monitors, the fission product water monitors, the stack gas and particulate monitors, and the nuclear incident monitors have a common bell located in the radiation monitoring instrument racks. A special alarm is provided on the console for the automatic power regulating system. It consists of two lights and a horn located on the console, and indicates a one inch deviation of the automatic control rod position from a preset position.

9.2.1.4 Control Rod Power Panel. The control rod power panel (No. 14 in Figure 9.1) distributes 120/240 volt, single phase, 60 cycle power to the main (safety) control rods and to the auxiliary control rods, including power to the clutches in the main control rod drive mechanisms. This is the only source of power available to the control rod drive units. The power panel is not connected to any source of emergency power.

9.2.1.5 Rod Control Rack. The rod control rack (No. 15 in Figure 9.1) contains the control rod drive motor phase shift networks, the terminal block panel for the rod drive wiring, the safety rod clutch power supply, and a control panel.

9.2.1.6 Emergency Power Rack. The emergency power rack (No. 16 in Figure 9.1) contains the following instrument emergency power elements: a battery charger, a solid-state inverter to produce alternating current from the batteries, and a set of flip-over relays.

9.2.1.7 Scram Relay Rack. The scram relay rack (No. 17 in Figure 9.1) contains the relays which break the safety rod clutch circuits upon a scram signal.

9.2.1.8 Setback Relay Rack. The setback relay rack (No. 18 in Figure 9.1) contains the setback relays and logic circuitry, the setback system relay power supply, and the scram relay power supply, the low-low logic panel, and an auxiliary logic panel.

9.2.1.9 Instrument DC Power Panel. The instrument DC power panel (No. 19 in Figure 9.1) supplies 125/250 VDC from the main battery bank in the fan house to the following equipment: HCe-101C siphon-break solenoid valve, HCe-102 depressurization solenoid valve, the nuclear incident alarm system circuitry, and the intercommunication rack.

9.2.1.10 Instrument AC Power Panel. The instrument AC power panel (No. 20 in Figure 9.1) supplies 120/208 VAC, 3 phase power to all instrument systems not powered from the control rod power panel or the instrument DC power panel. These include the leak detector system, the annunciator system, the process monitoring instrumentation, the nuclear instrumentation, the radiation monitor instrumentation, all electrically operated control valves, and the safety system circuits.

9.2.1.11 Fire Alarm Repeater Panel. The master fire alarm panel is located in the lobby of the south wing area. The repeater panel (No. 21 in Figure 9.1), duplicating the area alarm lights of the master panel, is located in the control room.

9.2.2 D₂O LEAK DETECTOR ALARM CABINETS. The heavy water leak detection system utilizes two alarm cabinets, one mounted on the equipment level near the water treatment area (column line 3, Figure 3.3) the other mounted on the operations level adjacent to the process equipment area (column line 11, Figure 3.6). Each cabinet contains eighteen indicator circuits with monitoring lights on the cabinet door. Also mounted on the cabinet door are the main power switch with monitoring pilot light, a test pushbutton for checking the indicator circuits, individual disconnect switches and alarm lights for each indicator circuit, and an "acknowledge" button to reset the alarm modules. The leak detector alarm cabinets are connected to the control room annunciator alarm panel, using one alarm point in the control room for each locally mounted cabinet.

9.2.3 HEAVY WATER TRANSFER SYSTEM. The heavy water transfer system (see Section 7.6.6) is operated from the vicinity of the transfer pumps GA-104A and GA-104B. These are located on the equipment level, near column line 2 (see Figure 3.3). The pumps are controlled by locally-mounted switches.

9.2.4 PRIMARY COOLANT PURIFICATION SYSTEM. The heavy water purification system (see Section 7.5) is controlled by manual operation of the valves in the system. These are located in trenches adjacent to the filter and ion exchange bed pits (see Figure 3.3).

9.2.5 CANAL COOLING SYSTEM. The canal cooling system is operated locally at the process units of the system. The canal circulating pump, GA-202, is controlled by a switch at the pump (see Section 7.8.5.2).

9.2.6 EMERGENCY GENERATOR CONTROL STATION. The emergency power generator, consisting of a propane gas-fueled engine directly connected to a 125 KW, 440 VAC, 3 phase generator is located outside the containment area on the equipment level. The generator starts automatically upon loss of normal power to the emergency control center. Shutdown of the unit must be done manually at the engine instrument panel located in the emergency generator room. All other controls are located at panels adjacent to the engine generator unit.

A separate annunciator panel is provided to monitor the emergency generator. This panel is located on the equipment level within the containment area, and is mounted on the wall of the generator room. The panel contains 5 annunciator points, one flasher section, one extra-loud horn and 3 control pushbuttons.

The annunciator is the "De-Line, Magna-Plac" as manufactured by the Scam Instrument Company. The operating sequence is identical to that of the control room annunciator. The alarm signals are for emergency generator startup, engine high water temperature, engine low lube oil pressure, and 2 spare points.

9.2.7 MANUAL SCRAM STATIONS. Seven manual scram stations are located throughout the building. Station 1 is on the console desk and consists of a momentary pushbutton and a lock-down key switch. Stations 2 through 7 consist of momentary pushbuttons and are located at each stairwell entrance (north and south sides of the building) on the operations level, experimental level and equipment level.

9.3 NUCLEAR INSTRUMENTATION

9.3.1 SAFETY CHANNELS 1, 2 AND 3

9.3.1.1 General Description. A block diagram of the nuclear safety channels is shown as part of the control and safety diagram, Figure 9.5. The

purpose of these channels is to provide setback or scram signals whenever the reactor period is too short and/or the reactor power is too high. Each channel receives signals proportional to the neutron flux from ionization chambers located in the reactor shield, and on the basis of these signals provides an output which is proportional to the sum of reactor power and inverse period. The point at which the safety channels will setback or scram the reactor is determined by a combined power and period signal such that the period required for trip is lengthened as the power level rises. Each channel also provides the operator with a "percent trip" signal which is displayed on a meter on the safety amplifier.

The setback and scram trip points are determined from the relations

$$\frac{P}{P_0} + \frac{\tau_0}{\tau} > 0.90 \quad \text{for setback}$$

$$\frac{P}{P_0} + \frac{\tau_0}{\tau} > 1.00 \quad \text{for scram}$$

where P is the reactor power, P_0 is the power level for scram at infinite period (50 MW), τ is the reactor positive period, and τ_0 is the positive period for scram at zero power (6 sec). Graphs of the limiting period as a function of reactor power, as determined from the above relation, are given in Figure 9.6. Curves for values of τ_0 of 3 and 10 sec are given as well as for 6 sec, to show the effect of varying this parameter. It will be noted from Figure 9.6 that the limiting power for setback is 45 MW (at infinite period), and that at the normal operating power of 40 MW a period shorter than 60 sec will cause setback.

Each of the three identical channels in the nuclear safety system is composed of three ionization chambers, a period amplifier, a safety amplifier, a dual voltage power supply unit, a constant voltage transformer, a setback relay, and a scram relay. Associated with all three channels are a safety monitor, a period recorder, and a power recorder. The relays are wired so that two of the three setback relays must open to produce setback control rod action, and two of the three scram relays must open to produce a rod scram. The safety and period amplifiers are "plug-in" units for quick replacement during servicing.

9.3.1.2 Safety Amplifier. The safety amplifier is the central element of each of the nuclear safety channels. A block diagram of a single safety channel, showing the safety amplifier functions is given in Figure 9.7, and the circuit diagram of the safety amplifier itself is given in Figure 9.8. The functions of the safety amplifier are summarized in the following tabulation.

The safety amplifier

- a) transmits a polarizing voltage of -400 VDC and a 1000 cps monitoring signal to the power level ionization chamber through the high voltage coaxial cable,
- b) receives from the power level chamber through the signal coaxial cable a negative direct current signal proportional to the neutron flux at the chamber and the 1000 cps monitoring signal,
- c) de-energizes the scram relay if the monitoring signal disappears due to failure of the chamber, the coaxial cables, or the high voltage supply,
- d) receives from the period amplifier a direct current proportional to the reciprocal of the reactor period,
- e) adds the inverse period and the power signals to produce a single trip signal which is displayed to the operator on a meter calibrated in "percent of scram trip",

- f) de-energizes the setback relay at a value of the trip signal equal to 90% of the scram trip value,
- g) de-energizes the scram relay at a value of the trip signal equal to 100% of the scram trip value,
- h) transmits signals proportional to the reactor power level and reactor period to the safety monitor for automatic sequential display on the power and period recorders, and
- i) receives manually-actuated test signals from the safety monitor which cause the setback and scram relays to operate, and which are used for checking and adjusting the power and period trip points.

The safety amplifier is a BNL design which employs both vacuum tubes and solid state rectifiers and transistors. It is identical to the safety amplifiers used on the other BNL reactors operated by the Reactor Division. A brief description of the circuitry of the amplifier follows, based upon Figure 9.8.

The power signal is fed to the safety amplifier at terminal (1) from the power level ionization chamber, and the inverse period signal is fed to the safety amplifier at terminal (2) from the period amplifier. The power level chamber is positioned so that the input current to the safety amplifier is $20\mu\text{a}$ at the normal reactor power of 40 MW. The input current from the power chamber is $25\mu\text{a}$ at 50 MW, the power level for scram at infinite period. The power chamber is a BNL type GN-3 (see Section 9.3.1.4). The input current from the period amplifier is zero at normal operating conditions with constant reactor power. The input current from the period amplifier is $25\mu\text{a}$ at a positive period of 6 sec, the period for scram at zero power.

The terminals (13) and (14) are the test signal input points for power and period, respectively, from the safety monitor. The recorder outputs for power and period, which go to the safety monitor, are terminals (4)-(5) and (4)-(3), respectively. The adjustments in the safety amplifier which determines the power for scram at infinite period (P_0), and the period for scram at zero power (τ_0), are made at the potentiometers P3 and P1, respectively.

The inverse period signal passes through an impedance-reducing stage (pins 1, 2, 3 of V-4) and is added to the power signal. The combined signal is fed to the tripping circuits through the cathode follower in the other half of tube V-4. The trip functions are performed by the vacuum tube circuits connected with tubes V-1 (pins 1, 2, 3) and V-2 for scram, and V-1 (pins 6, 7, 8) and V-3 for setback. These circuits are DC-coupled "bootstrap" amplifiers. The scram relay coil is connected between terminals (6) and (7), forming the plate load of V-2, and the setback relay coil is similarly situated between terminals (8) and (9) in the plate circuit of V-3. The relays are of the "normally-open" type, so that the scram and setback relay contacts are opened if the coil current fails (see Section 9.5 for a description of the relays). This configuration guards the safety amplifier against the common types of tube failures in the cathode follower stage of V-4, V-1 (both halves), V-2, and V-3, since all of these tubes must conduct normal currents to maintain the relay coil currents. The circuit is not completely guarded against a failure in the half of tube V-4 which is the period signal impedance reducer. Some protection is provided by the fact that this stage shares a common envelope and filament supply with the cathode follower half of V-4, and failure of these elements will cause a trip. This portion of the circuit is tested during every operating shift (every 8 hours).

The combined trip signal is displayed on a meter on the safety amplifier in terms of percent of the scram trip level. The meter current is derived from a suppressed-zero circuit consisting of the bias voltage network for V-1 (scram amplifier half) and the cathode follower output from V-4.

The power level ionization chamber and cable monitoring circuit obtains the 1000 cps monitoring signal from a transistorized blocking oscillator (Q-1, lower portion of Figure 9.8). The oscillator power is drawn from the 400 volt polarizing voltage supply for the power chamber, and thus automatically monitors the chamber voltage supply. The oscillator buffer amplifier (Q-2) is also a transistorized unit. The ionization chamber high voltage cable is attached to terminal (10), shown in the oscillator section of the diagram. The circuits monitoring the 1000 cps signal which returns from the chamber are shown in the upper portion of Figure 9.8. The monitoring signal is amplified in the two stages of V-5, rectified, and appears as a DC bias on the grids of V-6. The coil of the monitoring relay, C-1, forms the plate load for V-6. Again the relay is of the "normally-open" type, and the disappearance of the monitoring signal for any reason shuts off the coil current and allows the relay contact to open. The contacts of relay C-1 are in the cathode leg of the scram trip amplifier tube V-1, and upon opening cause the scram relay to open. The monitoring circuit is protected against plate power supply and tube conductive failures in the same manner as the scram and setback trip circuits.

The proper operation of both the power level and period ionization chambers and power supplies, and of the period amplifier is monitored for 20 minutes in each hour by the display of the power and period signals on recorders. A further protective interlock not shown in Figures 9.7 or 9.8 is provided by routing the scram relay coil current from the safety amplifier through the "plug-in" connector on the period amplifier. This insures that the period amplifier chassis cannot be removed without giving a scram trip on the safety channel in question. Removing the safety amplifier chassis from the instrument panel also causes a scram trip.

9.3.1.3 Period Amplifier. The functions of the period amplifier are to receive a direct current which is proportional to reactor power from ionization chambers and to transmit to the safety amplifier a direct current which is proportional to the reciprocal of the reactor period. The input current range is from 3×10^{-13} to 5×10^{-5} amperes. The period recorder range extends from -30 seconds (through infinity) to +3 seconds. Amplifier outputs are to the safety amplifier, the local period meter (infinity to +3 second scale), and to the local chamber current meter (3×10^{-13} to 5×10^{-5} ampere scale).

The period amplifier receives an input current which is the difference between the currents from two ionization chambers. One of the chambers is a BNL type GN-3 (Section 9.3.1.4) and the other is a BNL type G-1 (Section 9.3.1.5). The GN-3 chamber is boron-coated and is sensitive to both neutrons and gamma rays. The G-1 chamber is geometrically identical to the GN-3, but is not boron-coated and is thus sensitive only to gamma rays. The two chambers are positioned side by side in the same instrument tube in the shield and are polarized with voltages of opposite sign (the GN-3 is positive). The output currents of the two chambers are combined to obtain a difference current which is proportional to neutron flux only. The adjustment of the chamber outputs to remove the gamma-sensitive portion of the signal (compensation) is made by moving one chamber relative to the other. The normal input current to the period amplifier at 40 MW reactor power is 20 μ a.

The circuit diagram of the period amplifier is shown in Figure 9.9. The period output is obtained in the conventional way by amplifying a signal proportional to reactor power in a logarithmic amplifier and differentiating the log signal to obtain an inverse period signal (9.1, 9.2). The logarithm of the reactor power is obtained from a diode-connected type 5886 tetrode at the input of the period amplifier. This tube replaces the large resistor usually employed in an electrometer circuit. The tetrode is operated at reduced filament power

and, with the grids connected to the plate, gives a plate voltage proportional to the logarithm of the plate current for small currents.

An electrometer tube (a second type 5886 tetrode) amplifies the voltage across the "log" tetrode. A direct-coupled voltage amplifier employing a type 5654 tube amplifies the electrometer output. A type 12BH7A tube driven by the voltage amplifier and used as a cathode follower provides a low-impedance output for the differentiating and current-indicating circuits which follow (see Figure 9.9).

Within the normal operating range the amplifier is linear and the input diode characteristic is logarithmic. The output of the cathode follower will be equal to a constant voltage plus a voltage proportional to the logarithm of the input current. Converted into a current in the amplifier by means of a resistor, the logarithmic voltage directly operates the chamber current, or "Log N" meter mounted on the amplifier front panel.

Differentiation is approximated by a series R-C network. For the linearly increasing output voltage from the diode which occurs with a constant value of excess reactivity in the reactor, the total current in the differentiating capacitor will approach a constant value which depends upon the size of the capacitor. The capacitor current is passed through the period meter and causes deflections proportional to the inverse of the reactor period. The circuit is designed to produce full scale deflection of the period meter (located on the front panel of the amplifier) for a current corresponding to a period of 3 seconds. Two germanium diodes shunting the meter act to limit meter current and provide overload protection.

The logarithmic amplifier is of the DC adjustable feedback type. Calibration is required to make the chamber current meter indicate correctly. One calibration adjustment, marked " 5×10^{-11} adj", sets bias. Another, marked " 5×10^{-6} adj.", sets gain by varying the amount of feedback. Because the scale of the meter is logarithmic and the left-hand end is thus not zero, an unconventional calibration procedure is required. Two points on the scale are arbitrarily selected as calibration points. The linear characteristic of the amplifier assures that all other points on the scale are calibrated to a sufficient accuracy if these two points are correctly calibrated. Further, a third adjustment is provided to maintain zero potential at the input terminal when the current is at the lowest measurable level, in order to minimize the effects of leakage currents.

The period amplifier has certain dynamic response and static accuracy limitations, the most important of which are discussed here. The dynamic response of the logarithmic amplifier and inverse period circuitry is independent of the initial power level of the reactor (i.e., the initial chamber current), but is a function of the rate of change of power. Tests on a standard period amplifier, safety amplifier, and scram relay show that the trip time decreases with the input period (9.3), as shown below.

<u>Input Period</u>	<u>Time to Trip</u>
5.0 sec	1.0 sec
3.0	0.46
1.0	0.20
0.50	0.13
0.20	0.082
0.10	0.062

The "time to trip" includes the 5 millisecond opening time of the scram relay

(see Section 9.5.1.1). The tests were made at initial input currents of 3×10^{-13} , 1×10^{-11} , 1×10^{-8} , and 1×10^{-6} amperes, corresponding to reactor power levels of 0.6, 20, 2×10^4 , and 2×10^6 watts, respectively, with no change in the response times given above.

The total response time of the period channel is affected, at low power levels, by the charging time for the capacity of the coaxial cable from the ionization chamber. The effect of the cable capacitance is unimportant at power levels above 100 KW (9.3), as is shown in Table 9.3-1.

Table 9.3-1 Total Time to Open Scram Relay Contacts from Start of Power Rise, Period Channel Only

<u>Initial Reactor Power</u>		<u>3 sec Period</u>	<u>1 sec Period</u>	<u>0.1 sec Period</u>
20	watts	10 sec	4.4 sec	0.72 sec
200	"	4.0	2.3	0.49
2	KW	1.1	0.68	0.27
20	"	0.52	0.27	0.11
200	"	0.46	0.20	0.069
2	MW	0.46	0.20	0.062
20	"	0.46	0.20	0.062
40	"	0.46	0.20	0.062

The static accuracy of the period amplifier is limited by the tendency of the circuit to drift in calibration. When properly calibrated, the chamber current is indicated on the "Log N" meter with an accuracy of one percent of full scale deflection, which corresponds to an accuracy of 20% in chamber current. Experience with identical period amplifier units on the other BNL reactors show that the drift in the chamber current meter calibration (i.e., in the logarithmic amplifier) is about 2% of full scale per week. A daily calibration check is made on each of the period amplifiers, so that the amplifier drift between calibrations is about 0.3% of the full scale output signal.

The accuracy of the period output signal is about 3% for a constant period input, again from experience with identical amplifiers on the BNL reactors. In addition to the daily calibration check on the amplifier, the compensation of the ionization chambers which feed the period amplifier is checked and adjusted at every scheduled shutdown.

9.3.1.4 Gamma-Neutron Chamber GN-3. The gamma-neutron chamber, BNL type GN-3, is a boron-coated ionization chamber. The chamber is cylindrical, one inch in nominal diameter, and 14 to 18 inches long. The sensitive volume has a nominal length of 10 in., but may be increased by the manufacturer if necessary to achieve the required sensitivity. The case and electrodes are of .032-in.-thick aluminum, assembled by heliarc welding. Standoff steatite insulators support the signal and polarizing electrodes and glass to Kovar seals provide electrical connections to the signal and high voltage cables. The filling gas is a dry, 90% N_2 - 10% A mixture. The signal and polarizing electrodes are coated with 1 mg/cm² B-10, and the total B-10 content of the chamber is about 240 mg (9.4).

The minimum neutron sensitivity of the GN-3 chamber is 6.5×10^{-15} amp/unit thermal flux density, i.e., 6.5×10^{-15} amp/neut/cm²-sec. The gamma ray sensitivity is about 3.2×10^{-12} amp/r/hr. The chamber has a constant current response for constant neutron and gamma flux at polarizing voltages above 150 volts. The polarizing voltage normally supplied to the chamber is in the range 300-400

volts. The current response of the chamber is linear with neutron and gamma flux throughout the operating range from 50 to 8×10^9 n/cm²-sec in neutron flux, or 3×10^{-13} to 5×10^{-5} amp in chamber current. The polarity of the chamber current is the same as that of the polarizing voltage.

9.3.1.5 Gamma Chamber G-1. The gamma chamber, BNL type G-1, is an ionization chamber which does not have the B-10 coating of the GN-3 but is otherwise identical. The G-1 chamber responds to gamma rays but not to neutrons (other than through neutron capture gamma rays from the structural materials of the chamber). The gamma ray sensitivity of the chamber is 3.2×10^{-12} amp/r/hr, which is the same as that of the GN-3 (9.4). The G-1 chamber is used in conjunction with a GN-3 chamber with opposite polarization to obtain a compensated signal proportional to neutron flux.

9.3.1.6 Dual Voltage Power Supply. The dual voltage power supply (see Figure 9.7) provides polarizing voltages for the gamma and gamma-neutron chambers associated with the period amplifier. The gamma-neutron chamber is polarized positively and the gamma chamber negatively. Batteries supply 300 VDC, positive and negative, to the period system ionization chambers. Each of the six chambers is supplied by a 300 volt dry cell. At the normal chamber current of 20 μ amp the battery life is 3-6 months. Battery voltages are checked at each scheduled shutdown (3 week intervals), and batteries are discarded when the voltage at 1 ma load has decreased to 280 volts. Battery power supplies are necessary for the period system chambers to avoid spurious period trips from transients in the normal AC supply.

9.3.1.7 Constant Voltage Transformer. A constant voltage transformer for each channel (see Figure 9.7) provides power to the safety amplifier and to the period amplifier. In addition, the safety monitor is powered from the transformer for channel No. 1. The constant voltage transformer is manufactured by the Sola Electric Company. It has an input voltage range of 95 to 130 volts, and output voltage of 118 volts $\pm$ 1% with input voltage variations as great as 15%, and an output power of 250 VA. Power is fed to the constant voltage transformers from the Instrument AC Power Panel. No emergency power is provided for the nuclear safety channels (channels 1, 2, and 3).

9.3.1.8 Safety Monitor. The safety monitor is a switching unit for recorder signals from the three nuclear safety channels and for test signals to the safety channels (see Figures 9.7 and 9.5). The monitor functions are listed in the following tabulation.

- a. The safety monitor switches the reactor power and period signals from the three safety amplifiers to the power and period recorders in sequence. The switching occurs automatically at 20 minute intervals, starting on the hour, and the power and period signals from each of the three safety channels are displayed continuously for a 20 minute period.
- b. The safety monitor supplies manually controlled test signals which are fed into the safety amplifiers for use in adjustment of the period and power trip points on each safety channel.
- c. The safety monitor provides means for manually switching the reactor power and period recorders to any safety channel to check the power calibration or to read power and period from the channel.

9.3.1.9 Safety Channel Power and Period Recorders. The safety channel power and period recorders are Speedomax Type H Leeds and Northrup Instrument Co. strip chart recorders. The power recorder receives a 0 to 10 millivolt signal. The scale is linear, calibrated in megawatts reactor power from 0 to 60 MW.

The period recorder receives a ± 0.9 to ± 9.1 millivolt signal. The scale is calibrated in reactor period from -30 sec (through infinity) to $+3$ sec. Signals are fed to the power and period recorders from the safety monitor.

9.3.2 POWER LEVEL INDICATOR, CHANNEL NO. 4. The power level indicator is a linear reactor power channel which covers the full range from start-up to full power. The channel also provides differential power indication (see Figure 9.5). A signal proportional to neutron flux level is obtained from a compensated ionization chamber pair (one GN-3 and one G-1). The chamber current is amplified and is fed to two indicators on the control console, one for power level and one for differential power. Range switches for the amplifier and indicator lights to show the range are located on the console. The characteristics of the amplifier and the indicating instruments are given in Table 9.3-2. Power for channel No. 4 is obtained from the Instrument AC Power Panel.

Table 9.3-2 Channel No. 4 Amplifier and Indicator Characteristics

Amplifier

Amplifier model	Keithley Model 411 DC Amplifier
Input current range (as used)	10^{-11} to 10^{-4} amp
Accuracy	4%, current range 10^{-11} - 10^{-7} amp 2%, " " 10^{-7} - 10^{-4} amp
Response time (67% of final value)	1 sec, currents above 10^{-10} amp
Output	0 - 10 volts DC
Range changing	6 decades, remote switch and indicating lights on console

Indicators

Indicator model	Leeds and Northrup Speedomax, Type H, Circular Indicators
Type	null balance recording potentiometer
Input	0 - 10 volts DC
Scales	
Linear power	0 - 100, linear
Differential power	-5 to +5%, linear
Response time (full scale)	5 sec

9.3.3 THERMOPILE SYSTEM, CHANNEL NO. 5. The purpose of this system is to measure and provide an indication and record of the reactor power level in the operating range from a detector located below the top of the thermal shield. The channel is shown in block diagram form in Figure 9.5.

The detector consists of a helium-filled aluminum can containing a boron-coated thermopile manufactured by Minneapolis-Honeywell, Brown Instrument

Division, Type Y362731. The thermopile is formed of 20 Chromel-Alumel thermocouples in series. The thermocouples are arranged with all the hot junctions at the center of a radial array and the cold junctions on a circle surrounding the center. The hot junctions are imbedded in a globule of boron. Both hot and cold junctions are heated by the gamma ray field at the detector location, but this heating effect is approximately the same at both types of junction and is cancelled out of the signal from the thermopile. The hot junctions are also heated by neutron capture in the boron, which gives energetic alpha particles. The range of the alpha particles is short compared to the size of the boron globule, and the energy appears as heat. The equilibrium temperature of the boron, and of the hot junctions, is proportional to the neutron flux level at the thermopile.

The thermopile is mounted on a water-cooled shield plug in the upper thermal shield, and is placed just inside the thermal shield. The output of the thermopile is approximately 1×10^{-15} volts per unit thermal flux density. The thermal flux at the thermopile at normal operating power is in the range $2-5 \times 10^{12}$ n/cm²-sec, and the thermopile output signal is a few millivolts. A calibration shunt is provided on the output of the thermopile to correlate the signal and the recorder reading with the reactor power in megawatts.

The thermopile recorder is a Speedomax Type H Leeds and Northrup Instrument Co. strip chart recorder. The scale is linear and is calibrated to read 0-60 MW. The response of the recorder to a step change in input is six seconds for full scale travel. The input range is 0-5 millivolts.

9.3.4 POWER LEVEL CONTROLLER, CHANNEL NO. 6. The automatic power level control channel continuously adjusts one auxiliary control rod to maintain the reactor power at a constant value. The channel is shown in Figures 9.5 and 9.10. All controls for this channel are located on the console.

The system derives neutron flux information from two uncompensated ion chambers which supply signals to a Keithley amplifier, Model 411, identical to the amplifier described in Table 9.3-2. The amplifier has a remote range switch and indicating lights located on the console. The amplifier output goes to a power recorder and to the controller section of the channel.

The controller section of the channel consists of a power controller and a rod position controller. Both are DeVar indicating control units, Type C400. The power controller has a power set-point adjustment, a power indicating meter and a deviation meter. One output signal from the power controller is fed to an alarm module, the " ± 1 in. rod travel alarm". The alarm points correspond to ± 1 inch of deviation of the automatic rod from a preset point. The alarm is announced by lights and a bell on the console desk. The power controller has proportional, reset, and rate control actions and is arranged for cascade control of the rod position controller.

The rod position controller has a set-point adjustment for setting the lower limit of the control range for the channel. A rod position meter in the controller shows the actual position of the automatic rod so that the control range can be matched to the rod position prior to automatic operation. A deviation meter in the controller shows the difference between the actual rod position and the set-point. The position signal is taken from a potentiometer in the rod drive mechanism. One output signal from the controller is fed to the "auto mode reactivity interlock", which prevents automatic control unless the rod is withdrawn beyond a preset point. This interlock provides a means for limiting the total reactivity available in the automatic mode to less than one dollar. The rod position controller has proportional control action, with the main output

being a 1-5 ma DC signal which is fed to the rod positioner unit.

The rod positioner unit is a servo amplifier manufactured by the G. W. Dahl Co., Model A600. The positioner causes the automatic rod to move to a position proportional to the 1-5 ma control signal from the rod position controller. At 1 ma control signal the automatic rod is fully inserted, while at 5 ma the rod is fully withdrawn. The positioner unit receives an AC position signal from a differential transformer in the rod drive mechanism. The AC signal is converted to DC in the positioner unit and is compared to the control signal from the rod position controller. The output of the positioner unit is an "on-off" signal to the rod drive motor, which repositions the rod to eliminate the difference between the position and control signals.

Manual or automatic control of the rod is selected by the "auto-manual" switch located on the console. A setback signal will override an automatic or manual control signal to the rod, and the loss of signal from either controller will result in the rod being driven inward. The accuracy of the automatic control channel is limited by the flux-ion chamber current relation being linear only within 5%, by the amplifier accuracy of 2%, and the controller sensitivity of 1%. The short term accuracy is essentially limited by the incremental positioning accuracy, which is better than 1%.

9.3.5 COUNTING RATE SYSTEM, CHANNELS 7 AND 8. Two identical channels of the count rate type are used as start-up channels and for very low level operation. The channels are independent of one another except for a common power source. The channels operate on pulses from fission chambers. The chamber installation is described in Sections 9.3.6 and 9.3.7. A preamplifier for each channel is located at the reactor shield face for signal transmission to the control room. A count rate meter for each channel is located on the console. The count rate circuit chassis are mounted in instrument panels 8 and 10 in the control room.

A block diagram of the counting rate channels is shown in Figure 9.5. A single ground path is utilized to the preamplifier and detector to minimize noise problems. External signal and chamber polarizing voltages are carried via coaxial cables. The individual component specifications are given in Table 9.3-3.

Table 9.3-3 Channels 7 and 8 Component Specifications

Detectors:

Westinghouse fission chamber, Model WL8073
 Sensitivity: $0.7 \text{ (counts/sec)/(n/cm}^2\text{-sec)}$
 Neutron flux range: $0\text{--}1 \times 10^5 \text{ n/cm}^2\text{-sec}$
 Output pulse amplitude: 2×10^{-4} volts (unloaded)
 Output pulse rise time: 2×10^{-7} seconds
 Maximum operating temperature: 300°F .

Chamber Positioner:

Manufacturer: Bunnell Mechanical Tool Co.
 Positioner range: 36-48 inches
 Positioner speed: 24 in./min.
 Position indicator: Synchrogenerator
 Control: Push button switches on console

Preamplifier:

Design and manufacture: Milletron, Inc.

Table 9.3-3 (cont'd)

Circuit: Transistorized, with a common base input and direct coupled emitter output
 Voltage gain: 70 at .5 μ sec pulse width
 Noise: <1 millivolt

Main Amplifier:

Model: Hammer N 301
 Gain: 2×10^4 (without delay line clipping)
 Gain stability: 0.27% with line voltage change
 Gain drift: 0.25% per day
 Hum and Noise: 2 volts peak-to-peak

Log Count Rate Chassis:

Designer and manufacturer: Milletron, Inc.
 Range: 1 to 10^5 cps
 Circuit: Logarithmic diode with DC amplification
 External circuit: Ratemeter, repeated on console

9.3.6 DETECTOR LOCATIONS. The ionization chambers and fission chambers for the nuclear instrumentation are located in vertical tubes which terminate in the reactor shield at points just outside the thermal shield at the level of the core. The instrument tubes are 4.5 in. ID and contain a bottom vent connected to the building exhaust air system to provide cooling air flow inside the tube. A total of ten tubes are provided, designated "I-1" through "I-10". The assignment of chambers to these tubes is shown in Table 9.3-4 along with the reference elevation from core centerline to the centerline of the instrument port.

9.3.7 DETECTOR POSITIONING DEVICES. Manual positioners are provided on all instrument tubes containing ion chambers. The chambers are positioned with these devices to obtain the correct chamber currents at full power. The two fission chambers have motorized positioning devices which are controlled from the console. The remote-control positioners permit the operator to withdraw the fission chambers to regions of lower neutron flux during full power operation. The arrangement of the instrument tubes and the positioning devices are shown in Figure 9.11.

The upper end of each instrument tube provides a stepped section for shielding purposes. The chamber positioning mechanism and coaxial cable connections for the detector cables are located at the top of each tube. RG-58/U coaxial cable is used for all detectors. Steel conduit is used for electromagnetic shielding of the cables. Approximately 30 ft of cable is required in the instrument tube with another 75 ft of cable in the external run to the control room. The thermopile unit is located inside the thermal shield. The thermopile and its support plug are reached from the reactor pit during shutdowns.

The location of the instrument tubes in the region of the core midplane reduces the perturbation of the neutron flux at the ionization chambers due to control rod motion. It is expected that during normal reactor shutdowns the neutron level at the ionization chambers from the photoneutron reaction in heavy water will be sufficient to provide "on-scale" readings of the power indicating channels.

9.3.8 NUCLEAR INSTRUMENTATION FOR INITIAL LOADING AND START-UP. During the initial start-up, as well as in all start-ups following a complete refueling

Table 9.3-4 Nuclear Instrumentation Detector Locations

Instrument Tube	Elev. with Respect to Core Center	Chamber Type	Channel
I-1	-18 in.	GN-3	#1, safety amplifier
		GN-3	#6, power level control
I-2	+21	GN-3	#1, period amplifier
		G-1	#1, period amplifier
I-3	-9	fission chamber	#7, count rate
I-4	-18	GN-3	#2, safety amplifier
		GN-3	#6, power level control
I-5	+21	GN-3	#2, period amplifier
		G-1	#2, period amplifier
I-6	-9	GN-3	#4, power indicator
		G-1	#4, power indicator
I-7	+21	empty	Spare tube
I-8	-18	GN-3	#3, safety amplifier
I-9	+21	GN-3	#3, period amplifier
		G-1	#3, period amplifier
I-10	-9	fission chamber	#8, count rate

of the core an auxiliary neutron source will be utilized. The source to be used for the initial loading and the approach to critical will be a one curie radium-beryllium source. The source will be placed in one of the vertical irradiation facilities in the core.

For the initial start-up the normal instrumentation system will be utilized with some changes in chamber location. The ionization chambers for the safety channels, Channels 1, 2, and 3 will be moved to the thermal irradiation thimbles in the reactor vessel. This change will increase the neutron flux level at the safety channel detectors by a large factor, more than 10^4 for the maximum flux position, and will allow the reactor power level for scram at zero period to be set as low as a few kilowatts. The period trip will function as at normal power, with the period for scram at zero power set at 6 sec. By withdrawing the chambers upward in the thermal thimbles the flux level at the detectors will be decreased, for a given reactor power, and it will be possible to establish the reactor power for scram at infinite period at successively higher levels as the start-up experiments require increased power levels.

In addition to the normal reactor nuclear instrumentation, six low level linear power channels will be installed for the initial loading and the start-up experiments. Three of the low level channels will be of the counting type, and three will be ionization chamber channels. These channels will operate from detectors placed in the beam tubes, with the amplifier range switching equipment and the indicators and scalars located in the control room. The three low level

ionization chamber channels will be connected to the safety system in 1 out of 3 logic through spare scram relays.

9.4 NON-NUCLEAR REACTOR INSTRUMENTATION

The various process instruments are shown in the piping and instrumentation diagram, Figure 5.1. In general, the pressure, temperature, liquid level, 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 in appropriate cases are connected to the reactor safety system. As with the nuclear instrumentation, process instruments which are part of the reactor safety system are arranged for 2 out of 3 coincidence before a safety action is taken.

9.4.1 PROCESS INSTRUMENTATION IN THE SCRAM SAFETY SYSTEM. Of the eight non-nuclear reactor instrumentation systems which are part of the safety system, four are in the scram system and four are in the setback system. Those which will cause scram action are the reactor core differential pressure (high and low), the primary coolant flow (low), the primary system pressure (low), and the primary system coolant level (low). Each of the systems is wired in a 2 out of 3 coincidence circuit in order to insure adequate reactor protection together with reasonable freedom from accidental scrams from instrument malfunction. A block diagram of these instruments in the safety system is shown in Figure 9.5.

Tables 9.4-1a and 9.4-1b give a summary list of the scram system instruments, with normal process variable values and alarm and scram trip levels, for both normal operation and for single loop operation at reduced power. The single loop operation at the power level shown, 30 MW, is available as long as the cooling tower water can be delivered to the HFBR at or below 75°F. On an average of 25 days per year the atmospheric conditions can be expected to be such that the tower system supply water temperature will be above 75°F, and the single loop operating power will be accordingly reduced.

In the single loop 30 MW operation, the power for scram at infinite period (P_0 , see Section 9.3.1.1) is reduced to 37.5 MW. The positive period for scram at zero power (τ_0) remains at 6 seconds. With these settings the nuclear safety channels give a setback action at infinite period at 33.75 MW (rounded to 34 MW in the subsequent discussion). The positive period for setback at zero power remains at 6.7 seconds, as for full power operation. The nuclear instrument channels are included in Tables 9.4-1a and 9.4-1b to provide a complete listing of the scram system instruments.

9.4.1.1 Core Differential Pressure. The core differential pressure is monitored by the differential pressure transmitter PdRa-101 and two pressure switches, PdA-102 and PdA-103, which sense the reactor core differential pressure and actuate the scram circuits whenever 2 of the 3 instruments sense a high or low core pressure drop. Figure 9.14 shows the block diagram of these channels.

The sensing elements are two Barton Model 257 differential pressure switches and a DeVar Model 110 electronic differential pressure transmitter. The Barton "D/P" switches are equipped with stainless steel bellows which operate "high" and "low" adjustable "Milliswitches", and have ranges of 0-50 psi. The Milliswitch contacts are rated at 0.2 amp at 125 VDC. The contacts open to trip the safety system. The normal load on the Milliswitch contacts is the coil current of one scram relay, 0.06 amp at 74 VDC. The Barton D/P switches are located on the equipment level and provide a local indication of differential

Table 9.4-1a Instruments in the Scram System, with Set Points for Normal, 2-pump 40 MW Operation

Logic Grouping	Instrument No.	Service	Normal Reading	Scram Level	Pre-Alarm Level	Other Alarm Level	Other Trip Action
2/3	Nuc Chan #1	Neutron flux & period	40 MW ∞ sec	50 MW at ∞ sec 6 sec at 0 MW			Setback (Table 9.4-2)
	Nuc Chan #2	Neutron flux & period	40 MW ∞ sec	50 MW at ∞ sec 6 sec at 0 MW			Setback (Table 9.4-2)
	Nuc Chan #3	Neutron flux & period	40 MW ∞ sec	50 MW at ∞ sec 6 sec at 0 MW			Setback (Table 9.4-2)
2/3	PdRa-101	Core diff. pressure	33.5 psi	43 psi, HIGH 27 psi, LOW	41 psi, HIGH 29 psi, LOW		
	PdA-102	Core diff. pressure	33.5 psi	43 psi, HIGH 27 psi, LOW			
	PdA-103	Core diff. pressure	33.5 psi	43 psi, HIGH 27 psi, LOW			
2/3	FRa-100	Primary coolant flow	16,600 gpm	15,000 gpm	15,700 gpm		
	GA-101A	Primary pump power	on	off			Low-low logic
	GA-101B	Primary pump power	on	off			Low-low logic
2/3	LRa-101	Primary coolant level	69" D ₂ O (hot) 58" D ₂ O (cold)	41" D ₂ O	53" D ₂ O	74" D ₂ O, HIGH	29" D ₂ O, Low-low logic
	L1a-106	Primary coolant level	69" D ₂ O (hot) 58" D ₂ O (cold)	41" D ₂ O			29" D ₂ O, Low-low logic
	LA-102	Primary coolant level	69" D ₂ O (hot) 58" D ₂ O (cold)	41" D ₂ O			29" D ₂ O, Low-low logic
2/3	PRCa-101	Reactor He pressure	200 psig	180 psig	190 psig	220 psig, HIGH	120 psig, Low-low logic
	PA-102	Reactor He pressure	200 psig	180 psig			
	PA-103	Reactor He pressure	200 spig	180 psig			
1/8	Manual scram stations 1-7 and Lockdown switch	Manual reactor scram	closed	open			

Table 9.4-1b Instruments in the Scram System, with Set Points for Single Loop Operation at 30 MW

Logic Grouping	Instrument No.	Service	Normal Reading	Scram Level	Pre-Alarm Level	Other Alarm Level	Other Trip Action
2/3	Nuc Chan #1	Neutron flux & period	30 MW ∞ sec	37.5 MW at ∞ sec 6 sec at 0 MW			Setback (Table 9.4-2)
	Nuc Chan #2	Neutron flux & period	30 MW ∞ sec	37.5 MW at ∞ sec 6 sec at 0 MW			Setback (Table 9.4-2)
	Nuc Chan #3	Neutron flux & period	30 MW ∞ sec	37.5 MW at ∞ sec 6 sec at 0 MW			Setback (Table 9.4-2)
2/3	PdRa-101	Core diff. pressure	17 psi	27 psi, HIGH 13.5 psi, LOW	25 psi, HIGH 15 psi, LOW		
	PdA-102	Core diff. pressure	17 psi	27 psi, HIGH 13.5 psi, LOW			
	PdA-103	Core diff. pressure	17 psi	27 psi, HIGH 13.5 psi, LOW			
2/3	FRa-100	Primary coolant flow	11,800 gpm	10,500 gpm	11,000 gpm		
	GA-101A	Primary pump power	on (off)	off			Low-low logic
	GA-101B	Primary pump power	off (on)	off			Low-low logic
2/3	LRa-101	Primary coolant level	69" D ₂ O (hot) 58" D ₂ O (cold)	41" D ₂ O	53" D ₂ O	74" D ₂ O, HIGH	29" D ₂ O, Low-low logic
	LIa-106	Primary coolant level	69" D ₂ O (hot) 58" D ₂ O (cold)	41" D ₂ O			29" D ₂ O, Low-low logic
	LA-102	Primary coolant level	69" D ₂ O (hot) 58" D ₂ O (cold)	41" D ₂ O			29" D ₂ O, Low-low logic
2/3	PRCa-101	Reactor He pressure	200 psig	180 psig	190 psig	220 psig, HIGH	120 psig, Low low logi
	PA-102	Reactor He pressure	200 psig	180 psig			
	PA-103	Reactor He pressure	200 psig	180 psig			
1/8	Manual scram stations 1-7 and Lockdown switch	Manual reactor scram	closed	open			

pressure as well as the trip signals which are transmitted to the control room. The DeVar D/P transmitter, located on the equipment level, is an instrument of the temperature compensated strain gauge bridge type and has a differential pressure range of 50 psi with over-range protection to 3000 psig. It feeds an electronic signal to a recorder in the control room (DeVar Type R320) which contains the alarm assembly. The alarm assembly consists of a buffer module, a set-point trim potentiometer, an alarm module, and a relay. The relay contacts have the same rating and carry the same load as the Barton D/P switches. The Barton Milliswitches and the DeVar relay are connected to the scram circuitry so that whenever any 2 of the 3 units sense that the core differential pressure has exceeded prescribed limits a scram action will be initiated. The trip actions are indicated on the control room annunciator (see Section 9.2.1.3). In this application of sensing both a high and a low differential pressure, the strain gauge bridge will provide an alarm signal upon failure of any leg of the bridge.

The control room recorder (PdRa-101) associated with the DeVar D/P transmitter is supplied with two sets of high-low alarm modules; one set is used in conjunction with the scram trip and the other is used as a pre-alarm and will cause an alarm annunciator window to be illuminated before the core pressure drop reaches the point at which a scram will occur.

The normal core differential pressure at 16,600 gpm flow, as seen by the differential pressure instruments, is 33.5 psi. The D/P instrument pressure taps are located so that the 33.2 psi frictional pressure drop in the reactor vessel, plus 1.5 psi frictional pressure drop in the outlet line (20 in. Pl01) are measured. A static head difference of 1.2 psi, due to the elevations of the taps, is subtracted from the frictional pressure drop to obtain the D/P instrument readings. The high and low scram levels are set at 43 and 27 psi, and the pre-alarm points are set at 41 and 29 psi. The low differential pressure scram level of 27 psi corresponds to a flow rate of 15,000 gpm, which is the low flow scram level on the primary flowmeter (see Section 9.4.1.2).

For the 30 MW single loop operation at a primary flow rate of 11,800 gpm the D/P instrument normal reading is 17 psi. The high and low differential pressure scram levels are 27 and 13.5 psi, and the pre-alarm points are set at 25 and 15 psi. The low differential pressure scram level of 13.5 psi corresponds to a flow rate of 10,500 gpm, which is the low flow scram level on the primary flowmeter for single loop operation.

Some damping in the differential pressure instruments will be necessary to reduce the instrument response to hydraulic noise. A response time of the order of one second is expected for the damped instruments.

9.4.1.2 Primary Coolant Flow Rate. The primary coolant flow is monitored by the flow transmitter FRa-100 which senses the differential pressure across the venturi element in the reactor primary coolant line, and by two sets of auxiliary contacts in the primary pump switchgear. The flow signal from the venturi and the contacts in the switchgear of pumps GA-101A and GA-101B are utilized to form a 2 out of 3 coincidence circuit for the reactor coolant flow, causing a scram when the flow is below a preset limit. The block diagram of these channels is shown in Figure 9.12.

The sensing element for the venturi differential pressure is a DeVar Model 110 electronic differential pressure transmitter. The DeVar transmitter is of the same type as that used for the core differential pressure. The differential range is 250 inches of water, with over-range protection to 3000 psig. It is located on the equipment level adjacent to the shutdown heat exchanger area.

The venturi differential pressure signal is transmitted to a DeVar Type R320 recorder in the control room. A recorder-mounted DeVar alarm assembly, consisting of a buffer module, a set-point potentiometer, an alarm module, and a relay provides one of the three signals for the scram circuit, and also actuates a low flow pre-alarm signal on the control room annunciator panel. The relay contacts are rated at 0.2 amp at 125 VDC. The contacts open to trip the safety system. The normal scram relay coil current through the contacts is 0.06 amp at 74 VDC. The low flow scram and pre-alarm actuating contacts may be adjusted to permit reactor operation on either one or two primary pumps. This adjustment is done at the front of the instrument panel.

The auxiliary contacts in the primary pump switchgear, which trip the safety system when opened, are rated at 10 amp at 600 VAC. The scram relay coil current through the contacts is 0.06 amp at 74 VDC.

The normal coolant flow rate is 16,600 gpm. The low flow scram point is set at 15,000 gpm, and the pre-alarm point at 15,700 gpm. For single loop operation the normal flow is 11,800 gpm and the scram and pre-alarm points are set at 10,500 and 11,000 gpm. The response time of the flow and pump power sensing elements and circuitry is short compared to the times required to change the flow or start or stop a pump. The flow instrument and pump power monitors will follow conditions in the primary system with essentially no time lag.

9.4.1.3 Primary System Pressure. The primary system pressure is measured by the pressure control circuit PRCa-101 and the two pressure switches PA-102 and PA-103. The three instruments form a 2 out of 3 coincidence circuit causing a reactor shutdown if the pressure falls below a preset level. In addition, the pressure circuit PRCa-101 controls the helium gas pressure in the reactor vessel by means of two indicating controllers located on the console and two control valves, and provides a signal to the low-low pressure logic system (see Section 9.5.6). The control valves are located in the helium system on the operations level, one admitting helium and the other releasing helium from the vessel expansion space. Figure 9.13 shows the block diagram for the pressure channels.

The pressure sensing elements involved in the scram circuitry are a DeVar Model PG713TC strain gauge transducer and two United Electric Type J6 pressure switches. The DeVar transducer transmits a signal proportional to system pressure to a recorder in the control room. The United Electric instruments contain switches rated at 15 amp at 115 VAC which open to trip the safety system. A DeVar alarm assembly located in the recorder in the control room, and similar to that described in the core differential pressure circuits, is connected to the scram circuitry. The other two scram trips are taken from the pressure switches. Control room annunciator windows are illuminated to indicate which instruments have signaled a scram.

The alarm assembly in the recorder is equipped with one high and three low level switches. One low level switch is used in the scram system, one low level switch is used for the low-low pressure logic, and one high-low set of switches are used with the alarm annunciator. The low level switch of the latter pair is used as a pre-alarm to the scram system. Contact ratings are the standard 0.2 amp, 125 VDC, with the normal scram relay coil current load being 0.06 amp at 74 VDC.

The helium cover gas pressure at normal operating conditions is 200 psig for both single loop and full power operation. The scram level for all three pressure instruments is set at 180 psig. The low pressure pre-alarm from PRCa-101 is set at 190 psig, and the high pressure alarm from the same instrument is set at 220 psig. The low-low pressure trip point is 120 psig. The

response time of the pressure instruments is about 0.5 seconds or less. The strain gauge pressure transducer and the associated circuitry is very fast, and will follow the cover gas pressure with essentially no lag. The pressure switches are somewhat slower, but still are within the half second response time.

9.4.1.4 Primary Coolant Liquid Level. The reactor coolant liquid level is monitored by the two liquid level indicating transmitters LRA-101 and LIA-106, and by the liquid level indicating switch LA-102. These three units provide a 2 out of 3 coincidence circuit causing reactor shutdown if the coolant level falls below a preset low level. In addition, if the coolant level drops below the scram point and reaches a preset "low-low" level, the reactor flow reversal logic is activated, again in 2 out of 3 coincidence (see Section 9.5). The block diagram of these channels is shown in Figure 9.15.

The liquid level is measured by forcing helium into the reactor vessel at elevation 128 ft-1 in. through vertical pipes next to the vessel neck. The level instrument pipes extend above the vessel top and connect to one side of the differential pressure cells. The other sides of the D/P cells are connected to the cover gas volume. The helium pressure in the "bubbler" pipes then exceeds that in the cover gas volume by the liquid head above elevation 128 ft-1 in.

The instrumentation associated with these systems consists of two DeVar Model 110 differential pressure transmitters and their associated recorder alarm assemblies, and a Barton Model 257 indicating pressure switch. The DeVar transmitters are of the unbonded temperature compensated strain gauge bridge type. They transmit signals to a recorder and an indicator in the control room. The differential pressure range of the DeVar units is 112 in. of D_2O , with over-range protection to 3000 psig. The Barton pressure switch (LA-102) has a differential range of 112 in. of D_2O and over-range protection to 1000 psig. The Barton switch contains two low level Milliswitches which are opened to trip a scram relay and a low-low level logic relay. The indicating meter on the Barton switch is mechanically actuated by the differential gas pressure at the instrument, and does not depend upon an electrical power source.

The DeVar alarm assembly in the level recorder (LRA-101) has one high and three low level switches. The high level switch and one low level switch provide an annunciator window alarm for high and low liquid level. The low level alarm is set above the scram point and acts as a pre-alarm for scram. One alarm annunciator window is illuminated by both the high and low pre-alarm level switches. The other two low level switches are used to activate the low level scram logic and the "low-low" level flow reversal system logic. Both of these trip points are also signaled on annunciator window.

The DeVar alarm assembly on the indicating channel (LIA-106) contains two low level switches, one for the scram logic and one for the low-low level logic. Both of these trip points are signaled on annunciator windows. The trip contacts in the two DeVar units and the Barton switch are rated at 0.2 amp at 125 VDC. The loads on the scram and low-low level contacts are 0.06 amp, 74 VDC, and 0.1 amp, 24 VDC, respectively. The contacts open to trip the safety system.

The normal operating level for the primary coolant is 69 in. (elev. 133 ft-10 in.: elevations are obtained from the level instrument readings by adding the elevation of the entry points for the "bubbler" pipes to the vessel, 128 ft-1 in.). With no liquid having been added or removed, the 75°F shutdown level is 58 in. (elev. 132 ft-11 in.). The scram level is set at 41 in. (elev. 131 ft-6 in.), and the low level pre-alarm point is 53 in. (elev. 132 ft-6 in.). The high level alarm is set at 74 in. (elev. 134 ft-3 in.). The "low-low" logic

trip setting is 29 in. (elev. 130 ft-6 in.). The level settings are the same for both full power and single loop operation.

The response time of the "bubbler" gas system for sensing liquid static head, and of the differential pressure cells and circuits is short compared to the time required for appreciable changes in liquid level. The level monitoring instruments will, therefore, follow the liquid level with essentially no time lag.

9.4.2 PROCESS INSTRUMENTATION IN THE SETBACK SAFETY SYSTEM. The four process instrument systems which will cause setback action are the primary coolant core inlet temperature (high), the shutdown pumps (off), thermal shield coolant flow (low), and the biological shield coolant flow (low). These instrument systems are each arranged for 2 out of 3 coincidence for setback action except the shutdown pumps, where the logic is 2 out of 2. A block diagram of the setback safety system is shown in Figure 9.5.

Tables 9.4-2a and 9.4-2b give a summary list of the instruments in the setback system, with normal process variable values and alarm and setback trip levels for both full power and single loop operation. The nuclear instrument channels are included in Tables 9.4-2a and 9.4-2b to provide a complete listing of the setback system instruments.

9.4.2.1 Primary Coolant Core Inlet Temperature. The reactor vessel core inlet temperature is monitored by three independent temperature sensing circuits. The sensing elements are a resistance thermometer and two iron-constantan thermocouples. The resistance thermometer, TRa-101, is located in the outlet line of the primary heat exchangers and sees a temperature dependent on the combined flows from the exchangers. The temperature range of the thermometer is 20-70°C. The outlet water temperature from each of the heat exchangers is monitored by the two thermocouples, TIa-103 and TIa-104. Figure 9.12 shows the block diagram for these channels.

The resistance thermometer signal is fed to a recorder, DeVar Series R300, in the control room, and the two thermocouples are connected to two West "Gardsman" indicating pyrometric controllers in the control room. The West controllers, Model JL, have thermocouple burnout protection. The scales of the thermocouple indicators are calibrated from 0 to 150°C. The controllers are equipped with relay contacts rated 4 amp at 115 VAC and the trip point is indicated by a scale pointer. A DeVar alarm assembly with standard contact ratings (0.2 amp, 125 VDC) is located in the recorder for the resistance thermometer. The relay in the alarm assembly, in conjunction with the West "Gardsman" relays, are connected to setback relays which form a 2 out of 3 logic sequence to give setback action if the coolant temperature rises above a preset limit. The setback action continues until the instruments sense a temperature lower than the limit, at which time they will automatically re-set, and the setback action will cease. Appropriate control room annunciator windows indicate the condition of these instruments in the setback circuit.

In addition to the setback alarm point in TRa-101, additional switches in the DeVar alarm assembly give a low temperature alarm and a high temperature pre-alarm on the annunciator panel.

The normal coolant temperature at the core inlet is 120°F (~49°C) at full power. The setback point is set at 131°F (~55°C), and the pre-alarm for high temperature is set at 126°F (~52°C). For single loop operation the core inlet temperature is 123°F (~50.5°C). The setback point is set at 134°F (~56.5°C), and the high temperature pre-alarm is set at 129°F (~54°C). The low core inlet

Table 9.4-2a Instruments in the Setback System, with Set Points for Normal, 2-pump Operation at 40 MW

Logic Grouping	Instrument No.	Service	Normal Reading	Setback Level	Pre-Alarm Level	Other Alarm Level	Other Trip Action
2/3	Nuc Chan #1	Neutron flux & period	40 MW ∞ sec	45 MW @ ∞ sec 6.7 sec @ 0 MW			Scram (Table 9.4-1)
	Nuc Chan #2	Neutron flux & period	40 MW ∞ sec	45 MW @ ∞ sec 6.7 sec @ 0 MW			Scram (Table 9.4-1)
	Nuc Chan #3	Neutron flux & period	40 MW ∞ sec	45 MW @ ∞ sec 6.7 sec @ 0 MW			Scram (Table 9.4-1)
2/3	TRa-101	Core inlet D ₂ O temp.	49°C (~120°F)	55°C (131°F)	52°C (~126°F)	21°C (~70°F)	
	TIa-103	Core inlet D ₂ O temp.	49°C (~120°F)	55°C (131°F)			
	TIa-104	Core inlet D ₂ O temp.	49°C (~120°F)	55°C (131°F)			
2/2	GA-102A	Shutdown pump power	on (off)	off			Low-low logic
	GA-102B	Shutdown pump power	off (on)	off			Low-low logic
2/3	FRa-201	Lower thermal shield coolant flow	700 gpm	600 gpm	650 gpm		
	GA-201A	Thermal shield pump power	on (off)	off			
	GA-201B	Thermal shield pump power	off (on)	off			
2/3	FIa-203	Shutter coolant flow	100 gpm	70 gpm	85 gpm		
	GA-203A	Bio. shield pump power	on (off)	off			
	GA-203B	Bio. shield pump power	off (on)	off			

Table 9.4-2b Instruments in the Setback System, with Set Points for Single Loop Operation at 30 MW

Logic Grouping	Instrument No.	Service	Normal Reading	Setback Level	Pre-Alarm Level	Other Alarm Level	Other Trip Action
2/3	Nuc Chan #1	Neutron flux & period	30 MW ∞ sec	34 MW @ ∞ sec 6.7 sec @ 0 MW			Scram (Table 9.4-1)
	Nuc Chan #2	Neutron flux & period	30 MW ∞ sec	34 MW @ ∞ sec 6.7 sec @ 0 MW			Scram (Table 9.4-1)
	Nuc Chan #3	Neutron flux & period	30 MW ∞ sec	34 MW @ ∞ sec 6.7 sec @ 0 MW			Scram (Table 9.4-1)
2/3	TRa-101	Core inlet D ₂ O temp.	50.5°C (~123°F)	56.5°C (~134°F)	54°C (~129°F)	21°C (~70°F)	
	TIa-103	Core inlet D ₂ O temp.	50.5°C (~123°F)	56.5°C (~134°F)			
	TIa-104	Core inlet D ₂ O temp.	50.5°C (~123°F)	56.5°C (~134°F)			
2/2	GA-102A	Shutdown pump power	on (off)	off			Low-low logic
	GA-102B	Shutdown pump power	off (on)	off			Low-low logic
2/3	FRa-201	Lower thermal shield coolant flow	700 gpm	600 gpm	650 gpm		
	GA-201A	Thermal shield pump power	on (off)	off			
	GA-201B	Thermal shield pump power	off (on)	off			
2/3	FIA-203	Shutter coolant flow	100 gpm	70 gpm	85 gpm		
	GA-203A	Bio. shield pump power	on (off)	off			
	GA-203B	Bio. shield pump power	off (on)	off			

temperature alarm is set at 70°F (~21°C) for both cases. The response time of the temperature instruments is a few seconds.

9.4.2.2 Shutdown Pumps. The condition of the shutdown pumps is monitored by contacts in the shutdown pump motor switchgear. The contacts are rated at 5 amp, 600 VAC, and carry the normal setback relay coil current of 0.1 amp at 24 VDC. The contacts open to trip the safety system. If both pumps are shut down at any one time, a setback action will be initiated which will continue until one of the motors has been started. The condition, on or off, of each shutdown pump is indicated in the control room. Figure 9.14 shows the block diagram for this system.

9.4.2.3 Thermal Shield Coolant Flow. The thermal shield coolant flow is monitored by the flow transmitter FRA-201 and by auxiliary contacts in the thermal shield pump switchgear. The auxiliary contacts in the pump switchgear are rated at 10 amp, 600 VAC, and carry the normal setback relay coil current load of 0.1 amp at 24 VDC. The auxiliary contacts open with the pump breaker to trip the safety system. The flow transmitter signal and the signals from the two pumps are utilized to form a 2 out of 3 coincidence circuit which gives setback action for low thermal shield coolant flow or pump failure. The block diagram for these channels is shown in Figure 9.16.

The sensing element in the flow instrument FRA-201 is an orifice plate. The differential pressure across the orifice is detected by a DeVar Model 110 differential pressure transmitter. The transmitter is of the temperature compensated strain gauge bridge type. The differential range is 300 in. of water, and the cell has over-range protection to 3000 psig. The transmitter is located adjacent to the shutdown heat exchanger room on the equipment level and transmits a signal to a recorder in the control room. The recorder is a DeVar Type R320. A DeVar alarm assembly of standard contact rating (0.2 amp, 125 VDC) located in the recorder gives a setback signal and also a pre-alarm signal for low flow. The pre-alarm appears on the annunciator panel. The setback action, initiated by a low flow signal and loss of one pump, or by loss of both pumps, continues until the condition is remedied. After normal flow and pumping is established the setback action stops automatically. For operation of the thermal shield system on one pump, the setback logic becomes a 1 out of 2 system, as noted in Section 7.9.7.

The normal flow rate in the lower thermal shield is 700 gpm. The setback trip point on the flow instrument FRA-201 is set at 600 gpm, with the low flow pre-alarm set at 650 gpm.

9.4.2.4 Biological Shield Coolant Flow. The biological shield system coolant flow is monitored by the flow transmitter FIa-203 and auxiliary contacts in the biological shield coolant pump switchgear. The flow signal and signals from the two pumps are utilized in a 2 out of 3 coincidence circuit to cause setback action for low flow or pump failures. Figure 9.16 shows the block diagram for these channels.

The pump switchgear auxiliary contacts open with the pump breaker to trip the safety system. The auxiliary contacts are rated at 10 amp, 600 VAC, and carry the setback relay coil current of 0.1 amp at 24 VDC.

The sensing element in the flow instrument FIa-203 is an orifice plate. The differential pressure across the orifice is detected by a Taylor Instrument Company pneumatic flow transmitter located on the equipment level. The differential element is made of type 316 stainless steel, has over-range protection to 1500 psig, and has a pneumatic output of 3 to 15 psig. The pneumatic

signal is sent to two United Electric Controls Company Type J17A pressure switches (Model 142) located in the control room. The switches are independently adjustable. The switch contacts are rated at 15 amp, 115 VAC. One of the switches is used in conjunction with the setback circuitry and the other provides a pre-alarm. An annunciator window is illuminated when the biological shield coolant flow drops below a pre-set limit. Contacts in the switchgear are used to give signals to indicate the status of the pump motors on the annunciator panel.

The normal flow rate in the shutter circuit of the biological shield system, monitored by FIa-203, is 100 gpm. The setback trip point on FIa-203 is set at 70 gpm, and the pre-alarm is set at 85 gpm.

9.4.3 PROCESS INSTRUMENTATION IN THE ALARM SYSTEM. The process instrumentation in the alarm system includes all instruments which appear on the annunciator alarm panel, shown in Figure 9.3. A number of the alarm points have already been discussed in Sections 9.1 and 9.2. These are the pre-alarms for the scram and setback instruments, and several additional high or low alarms from the same instruments. The alarm points in this category are as follows (listed by instrument number):

PRCa-101, high and low	FRa-100, low
TRa-101, high and low	FRa-201, low
LRa-101, high and low	FIa-203, low
PdRa-101, high and low	

The remaining instruments on the alarm annunciator panel are discussed below. Table 9.4-3 provides a summary list of the "alarm only" instrumentation. The alarm points in Table 9.4-3 apply to both full power operation and to the single loop operation at 30 MW. In the single loop operation the "normal reading" of the secondary flow to the primary heat exchangers is reduced, as explained in Section 9.4.3.8.

9.4.3.1 Temperature Alarms. There are six "alarm only" coolant temperature points in the instrument system. All of these points are monitored by iron-constantan thermocouples in stainless steel wells. The thermocouples are connected to a twelve point recorder, Leeds and Northrup Co. Speedomax G, Model S, in the control room. The recorder scale is 0-100°C. Each temperature point has an adjustable alarm level in the recorder. The alarm signals from the recorder actuate the appropriate annunciator panel units. Table 9.4-3 lists the temperature points and the alarm levels. Figure 5.1 shows the location of these points in the process systems.

9.4.3.2 Primary-Secondary Pressure Difference. The pressure difference between the primary coolant and the secondary coolant in each of the main heat exchangers is measured and an alarm given if the difference drops below a pre-set value. The monitoring instruments are PIa-101A on the primary outlet and PIa-301A on the secondary inlet to cooler EA-101A and PIa-101B on the primary outlet and PIa-301B on the secondary inlet to exchanger EA-101B. The pressure difference between the instruments is the maximum difference for each exchanger.

The pressure transmitters in the primary system are manufactured by the U. S. Gauge Company, Model PlCO7N. They have a range of 0 to 400 psig and a proportional pneumatic output of 3 to 15 psig. The secondary system transmitters are the same U. S. Gauge Company instruments with a 0 to 100 psig range. The output of the secondary system transmitter feeds a Moore Products Company

Table 9.4-3 Instruments in Annunciator System on "Alarm Only"

<u>INSTRUMENT No.</u>	<u>SERVICE (T = temp., P = pressure, F = flow, L = level)</u>	<u>NORMAL READING</u>	<u>LOW ALARM LEVEL</u>	<u>HIGH ALARM LEVEL</u>
TRa-4	T, D ₂ O from exp. fac. cooler EA-102	49°C (~120°F)	21°C (~70°F)	55°C (131°F)
TRa-6	T, H ₂ O from thermal shield cooling circuits	52°C (~125°F)		55°C (131°F)
TRa-7	T, sec. H ₂ O from primary cooler EA-101A	19-49°C (~66-120°F)	16°C (~61°F)	
TRa-8	T, sec. H ₂ O from primary cooler EA-101B	19-49°C (~66-120°F)	16°C (~61°F)	
TRa-9	T, sec. H ₂ O supply from cooling towers	19-29°C (~66-84°F)	16°C (~61°F)	
TRa-11	T, H ₂ O from bio. shield cooling circuits	35°C (~95°F)		45°C (113°F)
PIa-101A PIa-301A	P, primary-secondary press. diff., EA-101A	195 psi (operating) 15 psi (shutdown)	10 psi	
PIa-101B PIa-301B	P, primary-secondary press. diff., EA-101B	195 psi (operating) 15 psi (shutdown)	10	
PIa-201	P, helium, low pressure supply manifold	50 psig	30 psig	
PIa-202	P, helium, high pressure supply manifold	300 psig	250 psig	
PA-104	P, reactor helium pressure	200 psig	120 psig	
PA-105	P, reactor helium pressure	200 psig	120 psig	
PA-301	P, instrument air	90 psig	75 psig	
FRa-101	F, D ₂ O, shutdown system to DC-101 in 6" P109	0 gpm (operating) 800 gpm (shutdown)	700 gpm (opens bypass valve at 300 gpm)	
FRa-300	F, sec. H ₂ O to primary coolers EA-101A & B	8000 gpm at 85°F 5000 gpm at 65°F	75% of normal	
FIa-10, 11, 12	F, D ₂ O to thimbles V-10, V-11, V-12	12 gpm	9 gpm	
FIa-13, 14	F, D ₂ O to thimbles V-13, V-14	17 gpm	13 gpm	
FIa-15, 16	F, D ₂ O to thimbles V-15, V-16	33 gpm	25 gpm	
FIa-204	F, H ₂ O to beam tube plug manifold	75 gpm	60 gpm	
FIa-205	F, H ₂ O to concrete cooling coil manifold	30 gpm	20 gpm	
FIa-301	F, sec. H ₂ O to shutdown cooler EA-103	150 gpm	100 gpm	
LIa-103	L, D ₂ O in exp. fac. cooler EA-102	Elev. 145'-6" (+0-3")	Elev. 144'-10"	Elev. 146'-4"
LIa-201	L, H ₂ O in thermal shield tank FA-201	Elev. 148'-7"	Elev. 147'-3"	Elev. 148'-11"
LIa-203	L, H ₂ O in bio. shield tank FA-203	Elev. 144'-11"	Elev. 144'-3"	Elev. 145'-10"
LA-307	L, H ₂ O in canal	Elev. 92'-6"	Elev. 92'-5"	Elev. 92'-7"
CRa-2	Primary D ₂ O electrical conductivity	1-2 µmho		5 µmho
RRa-302	D ₂ O leak to sec, shutdown cooler EA-103	bkgd count rate		4 x background
RRa-301A & B	D ₂ O leak to sec, primary coolers EA-101A & B	bkgd count rate		4 x background
CAB-1	D ₂ O leak to air, equipment level stations	no local alarms		≥1 local alarm
CAB-2	D ₂ O leak to air, operations level stations	no local alarms		≥1 local alarm
Door, North	Shield door position, north cell equip level	door closed		door open
Door, Cell A	Shield door position, cell A equip level	door closed		door open
Door, Cell B	Shield door position, cell B equip level	door closed		door open
48 VDC	48 VDC switchgear power supply	48 VDC	off	

pneumatic relay whose output is 3 to 6 psig. A United Electric Controls Type J6K differential pressure controller receives pneumatic signals from both the primary and secondary transmitters for each exchanger and causes the annunciator window to be illuminated for low differential pressure.

During normal operation the primary outlet (from the exchanger) pressure is 240 psig, as against the 45 psig secondary inlet pressure. On shutdowns the primary outlet pressure drops to the static head value of about 15 psig, while the secondary side of the exchangers is valved off and opened to atmospheric pressure. Thus the normal minimum pressure difference is about 15 psi, and occurs during shutdown. The low differential pressure alarm point is set at 10 psi.

9.4.3.3 Low-Low Reactor Pressure. The helium cover gas pressure in the reactor vessel is monitored by the pressure instruments PA-104 and PA-105, in addition to the instruments discussed in Section 9.4.1.3. The instruments PA-104 and PA-105 are part of the reactor low-low pressure main pump interlock system (see Section 9.5.6). In conjunction with PRCa-101, the instruments PA-104 and PA-105 form a 2 out of 3 logic system which shuts down both primary pumps (GA-101A and B) if the cover gas pressure in the reactor vessel falls below 120 psig. The PA-104 and PA-105 pressure switches are United Electric Co. Type J-6 units. The switch contacts are rated at 15 amp, 115 VAC. The switches open to actuate the pump shutdown. The low-low trip action of the switches is shown on annunciator panels in the control room. The third instrument in the logic system, PRCa-101, is discussed in Section 9.4.1.3. The block diagram of these channels is shown in Figure 9.13.

9.4.3.4 Helium Supply Manifold Pressures. Helium is supplied to systems throughout the plant from a high pressure supply and a low pressure supply (see Figure 3.27). The pressures of the two supply manifolds are monitored by P1a-201 in the low pressure system and P1a-202 in the high pressure system. The monitoring instruments are U. S. Gauge Co. pneumatic pressure transmitters with ranges of 0-500 and 0-100 psig for the high and low pressure manifolds, respectively. Each transmits a pneumatic signal of 3-15 psig to an indicating receiver gauge in the control room. The gauges are U. S. Gauge Co. Model 511S. The pneumatic signals are also transmitted to pressure switches which actuate the annunciator units. The switches are United Electric Controls Co. Type J6, Model 142. The normal manifold pressures are 300 psig and 50 psig for high and low pressure, respectively. The alarm points are set at 250 psig and 30 psig.

9.4.3.5 Instrument Air Pressure. Instrument compressed air at about 90 psig is supplied to various systems in the plant from compressors GB-301A and GB-301B. The air supply pressure is monitored by PA-301, a United Electric Controls Co. Type J6 pressure switch. The switch range is 0-100 psig. The low pressure alarm is set at 75 psig.

9.4.3.6 Shutdown System Coolant Flow. The flow rate in the shutdown cooling system is measured at an orifice plate in the inlet line to the reactor vessel (6 in. P109, Figure 7.1) by FRa-101. The orifice plate differential pressure range is 0-300 in. of water for a flow range of 0-1200 gpm. The differential pressure is sensed by a DeVar Model 110 differential pressure transmitter of the temperature compensated strain gauge bridge type. The transmitted signal is 0-40 millivolts. The signal is fed to a DeVar recorder in the control room and to an alarm assembly unit in the recorder. The alarm assembly has two independent low alarm relays. One of these, set at 700 gpm, controls the annunciator panel window. The other, set at 300 gpm, causes the bypass valve in line P180 (Figure 7.1) to open, bypassing water from the shutdown pump to the primary pump suction lines (see Section 7.1.1). Since the

shutdown system flow through line P109 is zero when the primary pumps are operating, the annunciator alarm for FRA-101 will be on during normal operation. The alarm set point of 700 gpm will be cleared only when the main pumps are down, and serves to monitor the nominal shutdown flow of 800 gpm during such times.

9.4.3.7 Shutdown Cooler Secondary Flow. The secondary water flow in the supply line to the shutdown heat exchanger is monitored to insure adequate water supply. The flow instrument is FIA-301, a Barton pneumatic pressure transmitter which transmits a 3-15 psig signal to an indicator and a pressure switch in the control room. The sensing element is an orifice plate on the secondary inlet side of the shutdown cooler, in line 4 in. WS-305. The orifice plate gives a 0-150 inch water gauge pressure differential for a flow range of 0-180 gpm. The indicator is a U. S. Gauge Co. Model 511S gauge with a 0-180 gpm scale. The pressure switch is a United Electric Controls Co. Type J6, Model 142 unit which controls the annunciator panel alarm. Normal flow in the system is 150 gpm and the alarm point is set at 100 gpm.

9.4.3.8 Secondary Flow to EA-101A and B. The secondary water flow rate in the supply line (20 in. WS-301, Figure 3.27) to the primary coolers EA-101A and EA-101B is monitored by a pitot tube sensing element and a Barton Instrument Corp. pneumatic differential pressure transmitter (FRA-300). The transmitter output is a 3-15 psig pneumatic signal which goes to a Taylor Instrument Co. recorder and a United Electric Controls Co. Type J6 pneumatic switch in the control room. Low flow in the secondary supply line is signaled by an annunciator window controlled by the switch.

The full power operating condition secondary flow in 20 in WS-301 is 8000 gpm at the maximum secondary water supply temperature of 85°F. Less flow is required as the secondary water temperature is lowered during cool weather, and at the nominal minimum supply temperature of 65°F only 5000 gpm are necessary. The alarm point of the pressure switch is adjustable over the full range of the transmitter signal, and is set at about 75% of the nominal secondary flow for the prevailing weather conditions. Thus, at 85°F secondary supply water temperature the alarm point is 6000 gpm, and at 65°F is 3750 gpm.

For single loop operation at 30 MW the required flow is 4000 gpm at 75°F. A lower temperature of the secondary water would require proportionately less flow. The alarm point of FRA-300 is set at 75% of the required flow, as in the full power case.

9.4.3.9 Irradiation Facilities Coolant Flow. The rate of coolant (D_2O) flow to each of the seven irradiation thimbles is monitored by flow instruments, and low flow alarms are displayed in the control room. The instruments are FIA-10 through FIA-16, monitoring coolant flow in thimbles V-10 through V-16, respectively. The flow sensing elements are orifice plates. The differential pressures across the orifice plates are 0-100 in. of water for flow ranges of 0-12 gpm in V-10, 11, and 12, and 0-35 gpm in V-13, 14, 15, and 16. The orifice plate differential pressure is detected by a Barton Instrument Corp. Model 257 indicating pressure switch for each thimble. The flow meter display for the thimbles is in the process area on the operations level. The switches control annunciator windows in the control room, one window for each thimble. The alarm points for low flow are set at 9 gpm for FIA-10, 11, and 12 (thimbles V-10, 11, and 12), at 13 gpm for FIA-13 and 14 (thimbles V-13 and 14), and at 25 gpm for FIA-15 and 16 (thimbles V-15 and 16).

9.4.3.10 Beam Tube Plug Coolant Flow. The rate of flow in the cooling water supply manifold for the beam tube plugs is monitored by FIA-204. The beam

tube plug manifold (3 in. TW223, Figure 7.14) is one of the sub-system lines in the biological shield cooling system. The sensing element is an orifice plate. The differential pressure across the orifice plate is detected by a Barton Instrument Co. Model 214 differential pressure transmitter with an input range of 0-100 in. of water and a pneumatic output signal of 3-15 psig. The output signal is transmitted to the control room to an indicating gauge (U. S. Gauge Co. Model 511S) and a United Electric Controls Co. Type J6 pressure switch. The switch controls the annunciator panel alarm for low flow in the plug coolant manifold. The normal flow rate in 3 in. TW223 is 75 gpm. The alarm point is set at 60 gpm.

9.4.3.11 Biological Shield Concrete Coolant Flow. The flow rate in the cooling water supply manifold to the cooling coils in the biological shield is monitored by F1A-205. The supply manifold (2 in. TW222, Figure 7.14) is part of the biological shield cooling system. The sensing element in F1A-205 is a Fisher-Porter variable area flow transmitter, Model 10A2152A-KF. The transmitter output is a 3-15 psig pneumatic signal which is sent to an indicating gauge (U. S. Gauge Co. Model 511S) and a pressure switch (United Electric Controls Co. Type J6) in the control room. The switch controls the annunciator panel alarm for low flow in the cooling coil manifold. Normal flow is 30 gpm and the alarm point is set at 20 gpm.

9.4.3.12 Experimental Facilities System D₂O Level. The liquid (D₂O) free surface in the experimental facilities cooling system is in the shell side of cooler EA-102. The level of this surface is monitored by L1A-103, a Mason-Neilan Co. Model 12823 float-type level transmitter with a 3-15 psig pneumatic output. The output signal is transmitted to an indicating gauge (U. S. Gauge Co. Model 511S) and a dual pressure switch (United Electric Controls Co. Type 17A) in the control room. The dual pressure switch controls a single annunciator panel window for both high and low level alarms. The experimental facilities cooling system is filled so that the liquid level during normal full power operation is at elevation 145 ft-6 in., i.e., the cooler shell volume of EA-102 (see Figure 8.9) is just filled. The "cold" level is then a few inches below this elevation. The high level alarm point is set at elevation 146 ft-4 in., which is just below (~1/2 in.) the invert of the reservoir tank. The quantity of D₂O which must be added to the system to raise the liquid level from the "hot" operating elevation to the high level alarm point is only 1.1 gal. The low level alarm is set at elevation 144 ft-10 in., 8 in. below the top of the cooler shell.

9.4.3.13 Thermal Shield System Water Level. The liquid level in the expansion tank FA-201 of the thermal shield cooling system is monitored by the instrument L1A-201 (see Figure 7.13). The instrument is a Mason-Neilan Co. Model 12823 float-type level transmitter with a 3-15 psig pneumatic output signal. The signal is transmitted to an indicating gauge (U. S. Gauge Co. Model 511S) and a dual pressure switch (United Electric Controls Co. Type 17A) in the control room. The dual pressure switch controls a single annunciator window for both high and low level alarms. The upper and lower pipe connections from FA-201 to the float tube of L1A-201 are at elevations 149 ft-4 in. and 146 ft-10 in., respectively. The normal operating level at full reactor power is at elevation 148 ft-7 in. The high and low trip levels are set at elevations 148 ft-11 in. and 147 ft-3 in., respectively.

9.4.3.14 Biological Shield System Water Level. The liquid level in the expansion tank FA-203 of the biological shield cooling system is monitored by the instrument L1A-203 (see Figure 7.14). The instrument is a Mason-Neilan Co. Model 12823 float-type level transmitter with a 3-15 psig pneumatic output signal. The signal is transmitted to an indicating gauge (U. S. Gauge Co.

Model 511S) and a dual pressure switch (United Electric Co. Type 17A) in the control room. The dual pressure switch controls a single annunciator window for both high and low level alarms.

The pipe connections to the L1a-203 float tube are at elevations 143 ft-8 in. and 146 ft-2 in. The instrument mid-range point, and the normal operating level at full reactor power is at elevation 144 ft-11 in. The high and low trip levels are set at elevations 145 ft-5 in. and 144 ft-3 in., respectively.

9.4.3.15 Canal Water Level. The canal water level is monitored by the instrument LA-307, with high and low level alarms transmitted to the control room to a single annunciator unit. Low pressure instrument air is bubbled into the canal through a dip tube, with the differential pressure between the dip tube air and the room atmosphere sensed by a dual pressure switch which controls the annunciator alarm. The normal canal water level is at elevation 92 ft-6 in., 6 in. below the equipment floor level. The high and low alarm points are set at elevations 92 ft-7 in. and 92 ft-5 in.

9.4.3.16 Primary Coolant Conductivity. The electrical conductivity of the primary D₂O is measured at the inlet to the purification system by the instrument CRa-2. An Industrial Instruments Corp. Model CEL-1 insertion-type conductivity cell is flange-connected to the pipe 2 in. Pl20 (see Figure 7.6) and is the conductivity sensing element. The cell pressure rating is 300 psig and all metal parts in contact with the D₂O are of type 316 stainless steel. The output signal from the cell is connected to one point (No. 2) of a Leeds and Northrup Co. Speedomax Type G eight-point recorder in the control room. The instrument scale is 0-20 micromho. An alarm point switch is provided for point No. 2 in the recorder, and controls the annunciator panel alarm for high conductivity on CRa-2. The normal primary D₂O conductivity reading is 1-2 micromho. The alarm point is set at 5 micromho.

9.4.3.17 D₂O Leakage. There are four alarm units in the annunciator system for D₂O leakage. Two of these units cover leakage from the primary system to the secondary system in the heat exchangers, and two units cover D₂O leakage from the primary and experimental facilities system to the building environment. The primary-secondary leakage in heat exchangers is monitored by the instruments RRa-302 on the shutdown cooler, EA-103, and by RRa-301A and RRa-301B on the main coolers, EA-101A and EA-101B. The instruments are described in Section 9.6, Radiation Monitoring Systems. On the annunciator panel, RRa-301A and RRa-301B are combined in a single alarm unit while RRa-302 controls a single unit.

The annunciator alarms for D₂O leakage to the building environment are controlled by the two leak detector systems, one on the equipment level and one on the operations level ("CAB 1" and "CAB 2"). These systems detect liquid leakage by the conductivity change in an absorbent material between two metal conductors. Wetting the absorbent material completes an electrical circuit and sounds a local alarm at the control cabinets for the systems. Each system has 18 points at flanges, valves, pumps, and instrument connections. The control room annunciator unit for each system alarms if one or more local detection points indicate leakage.

9.4.3.18 Shielded Cell Doors. The positions of the shielding doors to the three primary system process cells on the equipment level (see Figure 3.3) are monitored by switches which control alarm units on the annunciator panel. The switches give an alarm when the doors are in any position except fully closed.

9.4.3.19 Switchgear Power, 48 VDC. The 48 VDC switchgear power supply is monitored by a relay which gives an alarm on the annunciator panel in the event

of failure of the supply.

9.4.4 "NON-ALARM" PROCESS INSTRUMENTATION

9.4.4.1 Temperature Instrumentation. Temperature instruments which do not give alarm, setback, or scram actions are listed in Table 9.4-4. The instruments designated "TR" or "TI" in Table 9.4-4 have recorder or indicator displays in the control room. The "TG" instruments are local gauges of the bimetallic dial type, with cases and stems of stainless steel. The TG-5-50 gauges have a 0-50°C scale, while the TG-5-100 gauges have a 0-100°C scale. Both types are mounted in 5 in. deep stainless steel wells.

The instrument TR-102 is a resistance bulb thermometer whose output signal is transmitted to a two-point recorder in the control room. The other point on this recorder is TRA-101 (see Section 9.4.2). The recorder scale is 20-70°C. The temperature signals from TRA-101 (reactor coolant inlet temperature) and TR-102 (reactor coolant outlet temperature) are also supplied to the power calculator circuit, XR-100, as shown in the block diagram in Figure 9.12.

Table 9.4-4 Temperature Instrumentation, "Non-Alarm"

<u>Instrument No.</u>	<u>Service</u>	<u>Reference Figures</u>
TR-102	D ₂ O from reactor vessel	5.1, 5.2
TI-10	D ₂ O from thimble V-10	5.1, 8.9
TI-11	D ₂ O from thimble V-11	5.1, 8.9
TI-12	D ₂ O from thimble V-12	5.1, 8.9
TI-13	D ₂ O from thimble V-13	5.1, 8.9
TI-14	D ₂ O from thimble V-14	5.1, 8.9
TI-15	D ₂ O from thimble V-15	5.1, 8.9
TI-16	D ₂ O from thimble V-16	5.1, 8.9
TI-17	D ₂ O to EA-103, shutdown cooler	5.1, 7.1
TI-18	D ₂ O from EA-103	5.1, 7.1
TI-19	Sec. H ₂ O from EA-103	5.1
TR-1	Helium to recombiner, H-104	5.1, 7.5
TR-2	Helium from recombiner, H-104	5.1, 7.5
TR-3	D ₂ O to EA-102, exp. fac. cooler	5.1, 8.9
TR-5	H ₂ O to thermal shield (EA-201 outlet)	5.1, 7.13
TR-10	H ₂ O to shutters	5.1, 7.14
TG-5-50	3 points, EA-202	5.1
TG-5-50	3 points, EA-203	5.1
TG-5-50	Air Cond. cooling water outlet	6.1
TG-5-100	Sec. H ₂ O from EA-201	5.1
TG-5-100	Sec. H ₂ O from EA-102	5.1, 8.9
TG-5-100	Sec. H ₂ O from cooling towers	3.27

The temperature points TI-10 through TI-19 are iron-constantan thermocouples mounted in wells. The thermocouple signals are transmitted to a 24-point self-balancing electronic temperature indicator in the control room. The instrument scale is 0-100°C. The 14 unused points in the indicator are available as spares, or for additional temperature points.

The temperature points TR-1, 2, 3, 4, and 10 are iron-constantan thermocouple mounted in wells. The thermocouple signals are transmitted to a 12-point Leeds and Northrup Co. recorder in the control room. The other temperature points in this recorder (TRa-4, 6, 7, 8, 9, and 11) are discussed in Section 9.4.3.1. One point in the recorder remains open as a spare. The recorder scale is 0-100°C.

9.4.4.2 Pressure Instrumentation. Pressure instruments which do not cause alarm, setback, or scram actions are listed in Table 9.4-5. The instruments designated "PI" in Table 9.4-5 have indicator displays in the control room (except PI-305). The instruments designated "PI-T" or "PI-P" have indicating gauges mounted at the measurement point. Because of the large number of "PI-T" instruments, individual gauge locations are not given for all of these entries in the table. Reference to Figures 5.1 and 3.27 will show the locations of most of these instruments. The scale range of the "PI-T" gauges is given by the numbers following the letter designation. Thus, "PI-T400" is a 0-400 psig gauge. A double number shows a compound scale, with inches of mercury vacuum and positive pressure in psig. Thus, "PI-T30-15" is a 0-30 in. Hg vacuum, 0-15 psig gauge. The "PI-T" gauges are of drawn type 316 stainless steel. The "PI-P" gauges are bronze bourdon spring types with type 316 stainless steel diaphragm seals.

Table 9.4-5 Pressure Instrumentation, "Non-Alarm"

Instrument No.	Service	Reference Figures
PI-102A	D ₂ O, discharge of GA-102A	5.1, 7.1
PI-102B	D ₂ O, discharge of GA-102B	5.1, 7.1
PI-114	Helium, cover gas in FA-101	5.1
PI-115	Helium, cover gas in FA-102	5.1
PI-302	Sec. H ₂ O, supply from towers	3.27
PI-303	Sec. H ₂ O, return to towers	3.27
PI-305	CO ₂ , reactor cavity	3.25
PI-T30-15	Air, cover gas in FA-201	5.1, 7.13
PI-T30-15	Air, cover gas in FA-203	5.1, 7.14
PI-P160	2 units, sec. H ₂ O at GA-301	6.1
PI-T60	7 units	5.1, 3.27
PI-T100	10 units	5.1, 3.27
PI-T160	5 units	5.1, 3.27
PI-T200	Helium, cover gas in EA-102	5.1, 8.9
PI-T300	2 units, D ₂ O, GA-103A & B discharges	5.1, 8.9
PI-T400	15 units	5.1

The instruments PI-102A & B on the shutdown pump discharge lines are type

Table 9.4-9 Instruments with Operative Trip Functions During Initial Loading and Zero Power Tests

<u>Instrument</u>	<u>Trip Function</u>	<u>Reference Table</u>
Nuclear Channel #1	Scram, setback	9.4-1, 2
Nuclear Channel #2	Scram, setback	9.4-1, 2
Nuclear Channel #3	Scram, setback	9.4-1, 2
Startup Channel #1	Scram	
Startup Channel #2	Scram	
Startup Channel #3	Scram	
LRa-101	Scram	9.4-1
LIa-106	Scram	9.4-1
LA-102	Scram	9.4-1
Manual Scram, Buttons 1-7	Scram	9.4-1
PIa-101A/PIa-301A	Alarm	9.4-3
PIa-101B/PIa-301B	Alarm	9.4-3
PIa-201	Alarm	9.4-3
PIa-202	Alarm	9.4-3
PA-301	Alarm	9.4-3
LIa-201	Alarm	9.4-3
LA-307	Alarm	9.4-3
CRA-2	Alarm	9.4-3
CAB-1	Alarm	9.4-3
CAB-2	Alarm	9.4-3
48 VDC	Alarm	9.4-3

9.4.6 EFFECT OF TRIP LEVELS ON CORE TEMPERATURE AND BURN-OUT MARGIN.

Tables 9.4-1 and 9.4-2 give summary lists of the scram and setback instrument trip levels. Since each process variable trip function operates independently of the values of the other variables, consideration must be given to the effect on core temperatures and burn-out ratios of combinations of process variable levels at the trip points. For the purposes of this analysis only the hot spot temperatures and heat flux conditions are considered. As pointed out in Sections 4.7.3 and 4.7.7, the fuel temperatures at the hot spot and the corresponding heat fluxes are calculated on the assumption that all possible adverse conditions occur simultaneously and at the same point in the core.

The four variables which affect the fuel temperature and the burn-out margin are reactor power, primary flow rate, core inlet temperature, and primary pressure. All possible combinations of these variables at the scram or setback level are shown in Tables 9.4-10a and 9.4-10b, with the resulting hot spot surface temperature and burnout ratio. Table 9.4-10a refers to full power operation and Table 9.4-10b to single loop operation. Two cases are shown in each table: the equilibrium core case at the beginning of a cycle, and the fresh

Table 9.4-10a Effect of Setback and Scram Trip Levels on Core Temperature and Burnout Ratio - Full Power Operation

Variables at Trip Level	1	2	3	4	FRESH CORE							EQUILIBRIUM CORE				
	Reactor	Primary	Core D ₂ O	Primary	Hot Spot	Hot Spot	Hot Spot	Hot Spot	Hot Spot	Burn-out	Burn-out	Hot Spot	Hot Spot	Hot Spot	Burn-out	Burn-out
	Power	Flow	Inlet	Helium	Pressure	D ₂ O Sat.	Surface	Bulk	Heat	Heat	Ratio	Surface	Bulk	Heat	Heat	Ratio
	MW	gpm	Temp. °F	Pressure psig	psia	Temp. °F	Temp. °F	Temp. °F	Flux Btu/ hr-ft ²	Flux Btu/ hr-ft ²	ϕ_{bo}/ϕ_{hs}	Temp. °F	Temp. °F	Flux Btu/ hr-ft ²	Flux Btu/ hr-ft ²	ϕ_{bo}/ϕ_{hs}
(normal)	40	16,600	120	200	187	376	359	177	1.51x10 ⁶	3.72x10 ⁶	2.46	344	173	1.32x10 ⁶	3.77x10 ⁶	2.86
1	45	16,600	120	200	187	376	383	183	1.70 "	3.64 "	2.14	367	179	1.49 "	3.69 "	2.48
2	40	15,000	120	200	193	379	377	183	1.51 "	3.36 "	2.23	360	178	1.32 "	3.42 "	2.59
3	40	16,600	131	200	187	376	370	188	1.51 "	3.59 "	2.38	354	184	1.32 "	3.64 "	2.76
4	40	16,600	120	180	168	367	359	177	1.51 "	3.55 "	2.35	344	173	1.32 "	3.60 "	2.73
1, 2	45	15,000	120	200	193	379	401	190	1.70 "	3.28 "	1.93	382	185	1.49 "	3.34 "	2.25
1, 3	45	16,600	131	200	187	376	393	194	1.70 "	3.51 "	2.06	377	190	1.49 "	3.56 "	2.40
1, 4	45	16,600	120	180	168	367	383	183	1.70 "	3.47 "	2.04	367	179	1.49 "	3.52 "	2.36
2, 3	40	15,000	131	200	193	379	387	194	1.51 "	3.25 "	2.15	370	189	1.32 "	3.30 "	2.50
2, 4	40	15,000	120	180	173	370	377	183	1.51 "	3.21 "	2.13	360	178	1.32 "	3.27 "	2.48
3, 4	40	16,600	131	180	168	367	370	188	1.51 "	3.42 "	2.26	354	184	1.32 "	3.47 "	2.63
1, 2, 3	45	15,000	131	200	193	379	411	201	1.70 "	3.16 "	1.86	392	196	1.49 "	3.22 "	2.17
1, 3, 4	45	16,600	131	180	168	367	394	194	1.70 "	3.34 "	1.96	377	190	1.49 "	3.40 "	2.29
1, 2, 4	45	15,000	120	180	173	370	401	190	1.70 "	3.13 "	1.84	382	185	1.49 "	3.19 "	2.14
2, 3, 4	40	15,000	131	180	173	370	387	194	1.51 "	3.09 "	2.05	370	189	1.32 "	3.15 "	2.39
1, 2, 3, 4	45	15,000	131	180	173	370	411	201	1.70 "	3.01 "	1.77	392	196	1.49 "	3.07 "	2.07

Table 9.4-10b Effect of Setback and Scram Trip Levels on Core Temperature and Burnout Ratio - Single Loop Operation

	1	2	3	4	FRESH CORE							EQUILIBRIUM CORE						
Variables	Reactor	Primary	Core D ₂ O	Primary	Hot Spot	Hot Spot	Hot Spot	Hot Spot	Hot Spot	Burn-out	Burn-out	Hot Spot	Hot Spot	Hot Spot	Burn-out	Burn-out		
at Trip	Power	Flow	Inlet	Helium	Pressure	D ₂ O Sat.	Surface	Bulk	Heat	Heat	Ratio	Surface	Bulk	Heat	Heat	Ratio		
Level	MW	gpm	Temp. °F	Pressure psig	psia	Temp. °F	Temp. °F	Temp. °F	Flux Btu/ hr-ft ²	Flux Btu/ hr-ft ²	φ _{bo} /φ _{hs}	Temp. °F	Temp. °F	Flux Btu/ hr-ft ²	Flux Btu/ hr-ft ²	φ _{bo} /φ _{hs}		
(Normal)	30	11,800	123	200	201.6	382	364	183	1.13x10 ⁶	3.85x10 ⁶	3.41	344	179	0.99x10 ⁶	3.91x10 ⁶	3.95		
1	34	11,800	123	200	201.6	382	389	190	1.28	"	3.76	"	3.02	368	185	1.12	"	3.40
2	30	10,500	123	200	204.8	383.6	383	190	1.13	"	3.41	"	30.02	361	185	0.99	"	3.50
3	30	11,800	134	200	201.6	382	374	194	1.13	"	3.71	"	3.28	354	190	0.99	"	3.75
4	30	11,800	123	180	181.6	373.5	364	183	1.13	"	3.74	"	3.31	344	179	0.99	"	3.84
1, 2	34	10,500	123	200	204.8	383.6	410	198	1.28	"	3.32	"	2.59	387	192	1.12	"	3.02
1, 3	34	11,800	134	200	201.6	382	399	201	1.28	"	3.62	"	2.83	378	196	1.12	"	3.28
1, 4	34	11,800	123	180	181.6	373.5	389	190	1.28	"	3.65	"	2.85	368	185	1.12	"	3.30
2, 3	30	10,500	134	200	204.8	383.6	393	201	1.13	"	3.28	"	2.90	371	196	0.99	"	3.37
2, 4	30	10,500	123	180	184.8	375	383	190	1.13	"	3.30	"	2.92	361	185	0.99	"	3.39
3, 4	30	11,800	134	180	181.6	373.5	374	194	1.13	"	3.59	"	3.18	354	190	0.99	"	3.69
1, 2, 3	34	10,500	134	200	204.8	383.6	420	209	1.28	"	3.19	"	2.49	397	203	1.12	"	2.90
1, 3, 4	34	11,800	134	180	181.6	373.5	399	201	1.28	"	3.50	"	2.73	378	196	1.12	"	3.18
1, 2, 4	34	10,500	123	180	184.8	375	410	198	1.28	"	3.20	"	2.50	387	192	1.12	"	2.91
2, 3, 4	30	10,500	134	180	184.8	375	393	201	1.13	"	3.17	"	2.81	371	196	0.99	"	3.26
1, 2, 3, 4	34	10,500	134	180	184.8	375	420	209	1.28	"	3.07	"	2.40	397	203	1.12	"	3.79

core case, again at the beginning of the cycle. In both cases the hot spot temperature decreases during the operating cycle. The burn-out heat flux is calculated on the basis of the bulk water temperature rise at the stated conditions in the hot channel (9.5).

9.5 SAFETY SYSTEM

The safety system includes the relays, power supplies, and circuits which translate the switch actions of the various nuclear and process instrument trip units into motion of the control rods or of other safety devices. For purposes of the discussion here the safety system is subdivided into the scram system, the setback system, the flow reversal system, and the interlocks on main pump starting, control rod withdrawal, and main pump operation at low pressure.

9.5.1 SCRAM SYSTEM. The scram system causes fast insertion of the main control rods and normal speed insertion of the auxiliary rods upon appropriate signals from the nuclear and process instrument channels. The instruments which may cause a scram are listed in Table 9.4-1 and are shown in Figure 9.5. The components of the scram system and their functions are described in the sections which follow.

9.5.1.1 Scram Relays. The scram logic relay is the Westinghouse Air Brake Company Model GX. The coil operating data and contact information for this relay are given in Table 9.5-1.

The four heavy duty, normally open contacts of each relay are used in the scram logic circuitry. The four contacts are divided into two groups, each containing two contacts wired in series. The "double break" series configuration of the contact groups increases the voltage rating of the relay. Each group of contacts is used in a separate leg of the logic array, as shown in Figure 9.17.

Table 9.5-1 Wabco Relay Model GX: Scram Logic Relay

Coil Ratings at 20°C:

Normal operating voltage	74 VDC
Normal operating current	0.06 amp
Maximum pickup voltage	48 VDC
Minimum drop out voltage	5 VDC
Maximum voltage without over-heating coil	90 VDC

Release Time	5 millisec
--------------	------------

Contact Arrangement:

Heavy Duty Contacts	10 amp, 80 VDC
2 Normally open	
2 Normally closed	
2 Transfer type	
(normally open to closed)	
Light Duty Contacts	.5 amp, 80 VDC
2 Normally open	
2 Normally closed	
2 Transfer type	
(normally open to closed)	

The mechanical construction of the relay is of the spring leaf type with two rows of contacts. This arrangement allows the two contacts connected in series in each leg of the logic array to operate independently of each other. In the event that one contact becomes welded closed, the other contact in that group will open normally. The heavy duty contacts in the relay are equipped with permanent magnets for arc blow-out. The leaf spring arrangement also provides a considerable opening force.

The contact and coil assembly is mounted on a heavy insulating board and is completely enclosed in a metal housing. Connections to the relay are made at lugs on the outside of the insulating board. The board has a window for viewing the armature and is marked with the contact configuration for identification. Auxiliary (light duty) contacts are utilized in each relay to control the scram annunciator window associated with the instrument channel which controls the relay.

9.5.1.2 Manual Scram Buttons. The manual scram buttons are heavy duty units of the momentary break type, rated for 10 amp, 600 VDC for use with magnetic motor starters. The contacts are open as long as the pushbutton is depressed. A spring recloses the contacts as soon as the button is released. There are seven manual scram buttons, as shown in Figure 9.17, all breaking the coil current to the scram relay K-109. In this service the buttons are interrupting 0.060 amp at 74 VDC. Each manual scram has an alarm window on the scram annunciator panel.

9.5.1.3 Console Lockdown Switch. The console lockdown switch is a heavy duty, two position, key operated switch rated at 10 amp. In the "off" position the key may be removed, thereby locking the switch in that position (i.e., with the switch contacts open). The lockdown switch contacts interrupt the coil current to scram relay K-109, in series with the manual scram buttons (see Figure 9.17). The circuit carries 0.060 amp at 74 VDC.

The lighted "Reactor On" signs located throughout the building are controlled by the console lockdown switch, and are on whenever the lockdown switch is in the "on" position.

9.5.1.4 Scram Relay Coil Power Supplies. The scram relay coils receive power from the three safety amplifiers for scram relays K-106A, 106B, and 106C, and from a separate DC source for the other scram relays (see Figure 9.17). As noted in Section 9.3.1.2, each safety amplifier in the three nuclear safety channels supplies the corresponding scram relay coil with power from an internal DC supply. Each of the three nuclear safety channels is supplied power from a different instrument AC power phase. The failure of an AC phase from the instrument power panel, or of the DC supply within the corresponding safety amplifier will de-energize only the scram relay powered from that amplifier.

The process instrument channel scram relay coils are powered from a separate DC power supply rated at 2.5 amp at 80 VDC. This power supply consists of two full wave solid state rectifier bridge circuits coupled through blocking diodes. Each rectifier circuit is connected to a different phase of the AC instrument power. Since each rectifier circuit is able to supply the total relay load, the supply will continue to operate even though one of the AC power phases is de-energized. A tapped transformer is provided at the input to each rectifier circuit to allow the selection of the proper supply voltage.

Each transformer is rated at 2.5 amp, 120 VAC. The bridge and separator diodes are type 1N1614, rated at 5 amp at 200 volts peak inverse voltage. The

16 scram relays connected to this power supply constitute a continuously connected load of .96 amp. The failure of the process instrument scram relay coil supply will de-energize the 16 relays connected to it, resulting in a reactor shutdown.

9.5.1.5 Relay Logic Circuitry. Each channel of nuclear or process instrumentation connected to the scram system controls the coil circuit of a single scram relay. When any of these channels reach the preset scram limits, the scram relay coils for those channels are de-energized and the corresponding relay contacts in the scram bus circuit are opened. The relays remain de-energized as long as the abnormal condition in the reactor system exists. The individual scram channel relays are grouped into logic blocks of 3 relays each. There are 6 such logic blocks in the scram system. The scram bus is wired through the contacts of the 3 relays comprising a logic block in such a way that two of the three relays must have their contacts open for the logic block to interrupt the scram bus circuit (see Figure 9.17). The 6 logic blocks are in series, so that the opening of any one block will interrupt the circuit.

The seven manual scram pushbuttons and the console lockdown switch operate a single scram relay, K-109. The contacts of this relay are in series with the logic blocks, as shown in Figure 9.17.

9.5.1.6 Scram Bus Circuit. The scram bus circuit is composed of a power supply, the scram relay contacts, the eight main rod clutches, an auxiliary relay (SR-7), and arc suppression components. The arrangement is shown in Figures 9.17 and 9.19.

Power is supplied to the scram bus from a power supply at 67.5 VDC. The supply consists of a transformer and a bridge rectifier circuit (Figure 9.19). The power supply is connected to a single phase of the 120 VAC building power. A failure of the normal AC power results in failure of the scram bus power and shutdown of the reactor. The silicon diodes used for the bridge rectifier circuit are type 1N250A, rated at 20 amp at 200 volts peak inverse voltage. A 5000 microfarad filter capacitor is connected across the output of the rectifier (not shown in Figure 9.19).

The eight main rod clutch coils are connected in parallel across the scram bus. The clutch coil characteristics are given in Table 9.7-3. The coils will be operated at a maximum current (cold coil) of 450 milliamperes per coil at 67.5 VDC. The total current required for the eight clutch coils is 3.6 amp cold and approximately 25% less when the coils are hot. Since the clutch coils represent a large inductive load, an arc suppression circuit is connected in parallel with the coils. The circuit consists of a 10 ohm resistor in series with a 50 microfarad capacitor.

The power to the clutch coils is supplied through each of the 2 out of 3 logic blocks and the manual scram relay contacts, as noted in the previous section. The response time of the safety system, measured from the time the scram relay coil circuits are broken to the time the scram relay contacts open to break the control rod clutch circuit is 5 milliseconds.

The operation of the scram bus circuit is as follows (see Figure 9.19). To operate either the main or auxiliary control rods the scram bus must be energized. When the bus is energized, power is supplied to the main rod clutch coils and the coil of relay SR-7 through the scram relay contacts and the contacts of relay SR-7. In order to scram the reactor, it is necessary to open or de-energize the scram bus, thereby de-energizing the rod clutches and relay SR-7. This is done by the opening of the relay contacts in series in the scram

bus (K-109), the opening of two of the three contact groups in any scram logic block, or the failure of the AC power or DC power supply.

Relay SR-7 is a WABCO Model GX relay (see Section 9.5.1.1). When the scram bus is opened this relay performs the following functions:

1. Energizes relay TD-1, an Agastat time delay relay used to provide a four minute delay in the reclosing of SR-7.
2. Opens contacts in series in the scram bus, thus prohibiting the energizing of the main rod clutch coils until the timing circuit closes.
3. Closes the contacts located in the primary pump switch-gear circuits, permitting the pumps (GA-101A and B) to be started.
4. Opens a contact in the setback bus de-energizing the setback relays and causing the auxiliary rods to be inserted.

The contacts of relay TD-1 are in parallel with the holding contact on SR-7. The closing of the time delay contact energizes relay SR-7, permitting the scram bus to energize providing the scram relays have also reset. This feature of the circuit requires the scram bus to be open for a minimum of four minutes from the time that relay TD-1 is energized. The delay permits the auxiliary rods to be automatically inserted to the "full-in" position each time the reactor is scrammed. A capacitor is placed across the coil of relay TD-1 to delay the opening of the relay until the normally closed contact of relay SR-7 has fully closed.

9.5.2 SETBACK SYSTEM. The setback system causes all control rods to be driven inward at normal speed upon appropriate signals from the nuclear and process instrument channels. The instruments which may cause setback action are listed in Table 9.4-2, and are shown in Figure 9.5. The components of the setback system and their functions are described in the sections which follow.

9.5.2.1 Setback Relays. Similar relays are used for the setback logic relays (K-201, etc., Figure 9.18) and for the setback control relays (RR1-1, etc., Figure 9.19). These relays are built to MIL SPEC 6106 and have the standard part number AN-3304-1. The relays are built by the Electro-Mechanical Specialties Company. They are hermetically sealed units with a four-pole double throw contact configuration rated at 3 amp, 100 VDC. The three relays that are energized through the nuclear channel safety amplifiers have coils rated at 115 VDC. The remainder of the relays have coils rated at 24 VDC. The coil current is 0.1 amp at 24 VDC. The relays close at 13.8 volts and release at 4.7 volts. The setback system relays have a release time of 3 to 4 milliseconds.

9.5.2.2 Setback Relay Power Supply. The setback relay coils receive power from the three safety amplifiers for the relays K-204A, 204B, and 204C, and from a separate DC source for the other relays. The DC source is a dual bridge solid state rectifier supply similar to that described in Section 9.5.1.4 for the scram relays. The supply is rated at 4 amp at 24 VDC. The bridge and separator diodes are type 1N1342 and are rated at 6 amp at 100 volts peak inverse voltage.

9.5.2.3 Relay Logic Circuitry. Each nuclear or process instrument channel in the setback system controls a single setback relay coil circuit. When any of these channels reach the preset setback limit the setback relays for those channels are de-energized and the contacts in the setback bus circuit are opened. The relays remain de-energized for as long as the abnormal condition in the reactor system exists. The individual setback channel relays are grouped into logic blocks of three relays each, with the exception of the shutdown pumps (GA-102A and B) which comprise a logic block of two relays. There are a total of 14 setback relays, grouped into four logic blocks of three each and one logic

block of two. The setback bus is wired through the contacts of the relays comprising a three relay logic block in such a way that two of the three relays must have their contacts open for the logic block to interrupt the circuit. The contacts of the relays comprising the two relay logic block have their contacts in parallel, and both relays must open to interrupt the circuit. The five logic blocks are in series, so that any one may interrupt the circuit. The arrangement of the setback logic blocks are shown in Figure 9.18.

9.5.2.4 Setback Bus Circuit. The setback bus circuit is composed of a power supply, six setback control relay coils, the setback logic relay blocks, and a contact of relay SR-7. The six setback control relay coils are connected in parallel across the setback bus. The power to the control relay coils passes through the setback logic blocks and a contact of relay SR-7 of the scram bus. The circuit is shown in Figure 9.19. The power supply for the control relays is described in Section 9.5.2.2. It is the same supply which powers the setback logic relay coils.

The loss of power to the setback control relay coils may be caused by the opening of the SR-7 relay (as in the case of scram action), the opening of a setback relay logic block, or the failure of the power supply.

The contacts of the setback control relays are wired into the rod drive motor circuits in such a way as to (1) prohibit outward motion of the main and auxiliary rods during setback action, (2) drive the main and auxiliary rods into the core regardless of any other control rod action (except scram) which might be called for during a setback action, and (3) prohibit the automatic power regulation system from controlling auxiliary rod No. 1 when setback action is required.

The action of each of the six control relays to accomplish the functions listed above is as follows:

RR1-1: The No. 1 control relay utilizes 4 normally closed contacts to cause main rods 1, 3, 5, and 7 to be driven inward.

RR1-2: The No. 2 control relay performs the same action as No. 1 for main rods 2, 4, 6, and 8.

RR1-3: The No. 3 control relay utilizes 4 normally closed contacts to cause auxiliary rods 1, 3, 5, and 7 to be driven inward.

RR1-4: The No. 4 control relay performs the same action as No. 3 for auxiliary rods 2, 4, 6, and 8.

RR1-5: The No. 5 control relay is used to change the No. 1 auxiliary rod from automatic to manual operation for setback action. Three normally open contacts disconnect the 3 leads from the automatic rod positioner unit to the "auto-manual" switch. The fourth contact of RR1-5 is a normally closed contact which returns the No. 1 auxiliary drive motor to the "manual" AC source for setback action.

RR1-6: The No. 6 control relay is used to disable the withdrawal signal from the main and auxiliary manual control switches. Two normally open contacts are used for this purpose.

The relay RR-2, with contacts between the "rod positioner" unit and the "auto-manual switch" in the control circuit for rod A-1, is the "auto mode reactivity interlock" relay discussed in Section 9.3.4.

9.5.3 FLOW REVERSAL SYSTEM. The flow reversal system is designed to permit natural convection cooling of the reactor core in the event of a failure of all four primary and shutdown pumps, or in the event of a leak in the primary system which drops the liquid level in the reactor vessel. A detailed discussion of the hydraulic characteristics of this system is given in Section 7.2. This section will describe the circuitry associated with the system, as shown on Figure 9.20.

The loss of power to the primary and shutdown pumps causes the following actions:

1. Reactor scram, either directly from a power loss to the rod drive clutches or as a result of the opening of the breakers for the pumps (GA-101A and B, GA-102A and B).
2. The depressurization valve HCe-102 is opened as a result of the opening of the pump breakers.
3. The flow reversal valves open after the pumps are turned off and coast down.

The dropping of the liquid level in the reactor vessel causes the following safety system actions, which follow a low liquid level alarm on the annunciator panel:

1. Reactor scram from the low level trip.
2. The siphon break valve HCe-101C and the four pump breakers (GA-101A and B, GA-102A and B) open as a result of the low-low level trip.
3. The depressurization valve HCe-102 is opened as a result of the opening of the pump breakers.
4. The flow reversal valves open after the pumps are turned off and coast down.

9.5.3.1 Low-low D₂O Level Logic. Three separate liquid level instrument channels monitor the reactor vessel D₂O level. Each channel controls the coil circuit of a single low-low level logic relay. The contacts of the three logic relays (K-107A, 107B, and 107C) form a two out of three logic block, as shown in Figure 9.20. This block is connected in series with the pump stop relay coils (K-108A and K-108B). The opening of the contacts of two of the three logic block relays opens the logic block and de-energizes the pump stop relay coils. The contacts of the pump stop relays are located in the trip coil circuits of the primary pump switchgear, in the run coil circuits of the shutdown pump switchgear, and in the control circuit to the siphon break valve.

The low-low level logic relays are 24 VDC coil hermetically sealed MIL SPEC 6106 relays identical to those described in Section 9.5.2.1. The pump stop relays, K-108A and K-108B, are manufactured by the Allen Bradley Company, Type E-220A. They have six sets of contacts each, two normally closed and four normally open. The contacts are rated at 10 amp for heavy duty service and are capable of breaking a 1.1 amp inductive load at 250 VDC. The relay coils operate at 0.5 amp, 24 VDC. The relay operating time is approximately 50 milliseconds. The 24 VDC power supply used to operate both the logic relay coils and the pump stop relay coils is a battery system, see Section 9.8.1.2.

9.5.3.2 Main Pump Switching. The normally closed contacts of the pump stop relays K-108A and K-108B are connected in series with the trip coils of the main pump 2300 volt, 3 phase switchgear. Each pump has two sets of normally closed relay contacts in series with the trip coils. The contact action is to close the trip coil circuit, energizing the trip coil and thereby unlatching the spring-operated 2300 volt breaker and allowing it to open. The pump stop relay

contacts are in parallel with the other switchgear contacts operating the trip coils. In series with each trip coil is one set of breaker auxiliary contacts that function to open the trip coil circuits after the breaker is opened. Thus, the contacts for the pump stop relays serve to close the trip coil circuits, but not to open them. The trip coil requires 12 amp at 48 VDC to operate.

Upon the opening of the main pump breakers, two additional auxiliary contacts within the breaker trip the scram relays associated with the two main pumps, K-103B and K-103C. Auxiliary contacts of these relays are located in the four out of four logic control block of the depressurization valve, HCe-102.

Should the 2300 volt power to the main pumps fail, the trip coils are energized by undervoltage protection relays (not shown in Figure 9.20), and the main pump breakers are unlatched. The breaker auxiliary contacts controlling the scram relay coils (K-103B and K-103C) open, as above, and the scram relay auxiliary contacts in the depressurization valve control circuit are opened. The undervoltage relays have a short (3 sec at zero voltage) time delay, so that the main pump breakers are not thrown open by momentary faults in the power line.

9.5.3.3 Shutdown Pump Switching. Normally open contacts of the pump stop relays K-108A and K-108B are utilized to open the 250 VDC shutdown pump breakers in the event of a low-low liquid level signal. The run coils of these breakers must be energized to keep the pumps in operation, therefore the normally open contacts are used in series with the start, run, transfer, and trip contacts of each breaker. The run coils require 0.5 amps at 250 VDC to operate. The contacts of the pump stop relays open to turn the pump motors off. Auxiliary contacts in each shutdown pump breaker operate the setback relay associated with each pump. Auxiliary contacts of these relays are located in the four out of four logic control block for the depressurization valve HCe-102.

Should the 250 VDC power to the shutdown pumps fail, the run coils in the pump breakers are de-energized and the breakers open. The run coils draw power across the same supply as the pump motors. The breaker auxiliary contacts controlling the setback relay coils (K-202A and K-202B) would open in this case, as above, and the auxiliary contacts of these relays in the depressurization valve logic block would open.

9.5.3.4 Depressurization Valve Operation. The depressurization valve, HCe-102, is a spring-diaphragm operated valve which opens upon loss of air pressure (see Section 7.2.3). It is a 1 in. stainless steel valve with a 300 psig rating. The manufacturer is the Fisher-Governor Company. The porting arrangement consists of a single seat which is closed by a downward motion of the plug. The valve seats on 3 psig pressure and is normally held closed by 15 psig air pressure.

The 15 psig air pressure is supplied through a 3-way solenoid-operated valve which is mounted on HCe-102. The solenoid valve is manufactured by the Automatic Switch Company (ASCO) and is Model No. 830059-R. It has 1/4 in. orifice diameters and the valve body is rated for 75 psig service. The solenoid operates at 125 VDC and requires 12.1 watts. The power is supplied from the 125/250 VDC fan house battery supply. The solenoid is de-energized to dump the air from the HCe-102 operator, allowing HCe-102 to open. The four primary coolant pumps must be shut down to de-energize the solenoid. The shutdown of the four pumps may be due either to the low-low level logic operating the pump stop relays, or to pump power failure. In either case, the four sets of parallel contacts of the pump scram and setback relays must open to de-energize the solenoid.

9.5.3.5 Siphon Break Valve. The siphon break valve, HCe-101C, is a 4 in. piston-spring-operated valve. The piston is 13 inches in diameter and has a 4 inch stroke. The spring is designed to hold the valve closed against a 300 psig head on either side of the seat. The valve has a 300 psig rating and is flange mounted. It has a single seated port and has a cast aluminum body. The valve is manufactured by the Fisher-Governor Company, and is Type 473-Z.

In normal operation a pressure of 100 psig of helium is maintained on both sides of the piston and the valve is held closed by the spring. The de-energizing of the solenoid control valve vents the spring side of the piston to atmospheric pressure, which allows the valve to open (see Figure 9.20). If helium pressure is lost from both sides of the piston, as would be the case for a failure of the gas supply, the siphon break valve is held closed by the spring.

The solenoid valve is a Automatic Switch Company (ASCO) 3-way valve, Type 8300, with a 125 VDC coil requiring 15 watts. It is a direct-acting solenoid with metallic seats. The solenoid valve has a maximum pressure rating of 600 psig and has 1/8 in. diameter ports. The DC power to operate the solenoid is derived from the 125/250 VDC fan house battery supply. The helium for the system is derived from the high pressure helium header via a 1/4 in. line, check valve, strainer, and pressure reducer. Tubing for the system is 1/4 in. diameter. The solenoid is energized through normally open contacts of the pump stop relays K-108A and K-108B.

9.5.4 PRIMARY PUMP START INTERLOCK. This interlock requires the reactor to be shut down in order to start either of the primary pumps GA-101A and GA-101B. The interlock prevents the sudden introduction of cold D₂O into the reactor vessel during reactor operation due to the starting of one of the primary pumps.

The SR-7 relay, which is a component of the scram bus, is de-energized when the control rod clutch coils are de-energized. The normally closed contacts of this relay are in series with the closing coil contact relays of the pump 2300 volt breakers. Two sets of contacts are arranged in series in each of the pump breaker control wiring circuits. When SR-7 is energized, as it is when the scram bus is energized, the contacts are open and the closing coils of the main pump breakers cannot be energized.

The main pump closing coil control circuit operates at 250 VDC. The SR-7 relay is identical to the scram relay described in Table 9.5-1, with heavy duty contacts rated at 10 amps at 80 VDC. The use of two sets of contacts in series permits these contacts to be used at the higher voltage.

9.5.5 CONTROL ROD GANG WITHDRAWAL INTERLOCKS. These interlocks prevent the withdrawal of any auxiliary control rod or the gang of auxiliary control rods when the main control rod selector switch on the console is in the "gang" position, and vice versa. The interlock is provided by contact No. 17 of the main rod SB-1 selector switch (RSM-17) in the "withdraw" circuit of the auxiliary rods, and contact No. 17 of the auxiliary rod SB-1 selector switch (RSA-17) in the "withdraw" circuit of the main control rods. The No. 17 contacts are open when the rod selector switches are in the gang position. The only exception to this interlock is the operation of auxiliary control rod No. 1, which may be withdrawn automatically when the main control rod selector switch is in the "gang" position. The arrangement of selector switch contacts RSM-17 and RSA-17 is shown in Figure 9.19. This interlock does not affect in any way the simultaneous insertion of any combination of main and auxiliary control rods.

9.5.6 LOW-LOW PRESSURE MAIN PUMP INTERLOCK. The low-low pressure interlock shuts down both primary pumps, GA-101A and GA-101B, if the primary system pressure falls below 120 psig in the helium cover gas space. The interlock is controlled by low-low pressure instruments (PRCa-101, PA-104, PA-105) which monitor the helium pressure and close contacts in the main pump breaker trip coil circuits when the pressure is too low. The circuits are shown in Figure 9.21 and the instruments are discussed in Section 9.4.3.3.

Three separate pressure channels monitor the reactor pressure, each channel controlling a single logic relay. The contacts of the three logic relays form a 2 out of 3 logic block connected in series with the pump stop relay coils (K-113A and K-113B). The opening of the contacts of two of the three logic block relays opens the logic block and de-energizes the pump stop relay coils. The normally closed contacts of these relays are located in series with the trip coils of the primary pump breakers. The trip coil action is the same as that which occurs on a low-low level signal, described in Section 9.5.3.2. The logic and pump stop relays are identical to those used in the low-low level system, and make use of the same power supply (see Section 9.5.3).

9.5.7 SAFETY SYSTEM DURING INITIAL LOADING AND START-UP. Many of the scram and setback logic blocks in the safety system will be by-passed during the initial loading and zero power tests to permit control rod motion with the primary system at zero flow and atmospheric pressure.

Table 9.4-9 in Section 9.4.5 lists the instrument channels which will have operative trip functions during the initial loading. Of the scram and setback channels only the three nuclear safety channels and the three liquid level channels will cause rod action. Both these sets of instruments operate in the scram system. The logic blocks in the scram bus circuit of the other instruments will be by-passed, as will all the setback logic blocks. There will, however, be three low level ion chamber channels added to the scram system, with a 1 out of 3 logic block. The nuclear channels are necessarily operative in the initial loading for the safety of the system. The liquid level channels are kept operative in this period only to protect the main rod drives against a dry scram.

The flow reversal system serves no useful purpose in plant protection as long as there is no substantial fission product inventory in the core to generate afterheat. The system will be by-passed to permit retention of a few pounds pressure of inert cover gas in the reactor vessel without the pumps operating, for such periods as the vessel may be closed.

The primary pump start interlock will be operative, since there is no reason to operate the pump breakers while the scram bus is energized.

The control rod gang withdrawal interlocks will be operative.

The low-low pressure main pump interlock will be by-passed, to permit short main pump starts to check the pump breaker operation without having 120 psig cover gas pressure. No extended running of the main pumps can be done, of course, without sufficient pressurization to prevent cavitation.

9.6 RADIATION MONITORING SYSTEMS

9.6.1 AREA RADIATION MONITORS. The area radiation monitoring system continuously monitors ambient gamma dose rates at specific locations within the reactor building. The system consists of ten stations located as follows:

EA-101A Primary Cooler Area, Equipment Level

EA-101B Primary Cooler Area, Equipment Level
 EA-103 Shutdown Cooler Area, Equipment Level
 Column 12C, Equipment Level
 Column 3C, Equipment Level
 Column 5C, Experimental Area
 Column 13C, Experimental Area
 Reactor Pit
 Southwest Area, Operations Level
 Northeast Area, Operations Level

The detectors are Tracerlab Type TA-6 halogen-quenched G-M tubes. Each detector assembly is equipped with an activity level indicator, alarm light, alarm buzzer, and remotely actuated operational check source. The detector range is .01 mr/hr to 10 r/hr.

The radiation levels at each detector are indicated in the control room by individual meter relay stations (Tracerlab Type TA-3). Alarm levels may be independently set over the entire detector range. When the measured activity exceeds preset levels, an alarm condition is indicated by visual and audible signals at both the detector and control room locations. The normal alarm setting is 10 mr/hr. A solenoid-operated check source, actuated from the control room, provides a means of periodically checking the operability of each detector and meter station.

9.6.2 SECONDARY WATER MONITORS. The secondary water is monitored for radioactive contaminants by four "on-line" instruments and by the analysis of samples taken every eight hours. The "on-line" instruments continuously measure and record the activities of the secondary effluents of each heat exchanger containing D₂O to provide an indication of leakage of D₂O into the secondary system. The four heat exchangers which contain D₂O are EA-101A and B, EA-103, and EA-102. The leak detector instruments for these exchangers are RRa-301A, RRa-301B, RRa-302, and RRa-303, respectively.

9.6.2.1 RRa-301A And RRa-301B. The sensing elements of RRa-301A and B are Tracerlab MD-5A gamma scintillation detectors mounted in 3 in. diameter stainless steel protective wells inserted in the 16 in. secondary effluent lines from EA-101A and B (see Figure 5.1). The detector consists of a 1-1/2 in. diameter by 1 in. long thallium-activated sodium iodide crystal and a Dumont 6292 photomultiplier. A concentration of $6 \times 10^{-6} \mu\text{C}/\text{cm}^3$ of 1.12 Mev gamma activity in the effluent water will produce a net signal of 250 cpm above a background reading of 250 cpm.

The output of each detector is indicated on a Tracerlab RM-20B log rate-meter and a DeVar multipen recorder, both located in the control room. A meter relay activates a control room alarm annunciator circuit when the indicated activity exceeds preset levels. The alarm setting is four times background. A solenoid-operated check source, actuated from the control room, provides a means of periodically checking the operability of each system.

9.6.3.2 RRa-302. The sensing element of RRa-302 is a Tracerlab MD-5B gamma scintillation detector housed in a 4 in. carbon steel flanged pipe "tee" liquid sampler assembly (Tracerlab MW-2P-V5). The detector assembly is mounted in the secondary water effluent line from the shutdown cooler, 4 in. WR305 (see Figure 5.1). The detector consists of a 1-1/2 in. diameter by 1 in. long thallium-activated sodium iodide crystal and a Dumont 6292 photomultiplier. A concentration of $6 \times 10^{-6} \mu\text{C}/\text{cm}^3$ of 1.12 Mev gamma activity in the effluent water will produce a net signal of 250 cpm above a background of 250 cpm. The detector output is indicated on a Tracerlab RM-20B log count rate meter and a

DeVar multipen recorder, both located in the control room. A meter relay actuates an alarm on the control room annunciator when the indicated activity exceeds a preset level. The alarm setting is four times background. A detector test source assembly, actuated from the control room, provides a test signal to check the system.

9.6.2.3 RRa-303. The instrument RRa-303 is similar to RRa-302, except that it is mounted in a 3 in. pipe "tee" assembly on the secondary effluent line from the experimental facilities cooler EA-102. The line number is 3 in. WR309 (see Figures 5.1 and 8.9).

9.6.2.4 Secondary Water Sample Analysis. Samples of the secondary effluent from the heat exchangers which contain D_2O , as well as other liquid samples from various plant operations are periodically analyzed for tritium content. The instrument which is used for the tritium analysis is a Packard Instrument Co. "Tri-Carb" liquid scintillation spectrometer. The instrument is located in the heavy water laboratory on the operations level of the building. Samples are prepared for the tritium analysis by filtering to remove opaque impurities, reduction to a standard sample volume by pipetting, and dilution with a standard volume of liquid scintillator solution. Once the samples are prepared and placed in the spectrometer, the operation of the instrument is automatic. The samples are kept in a refrigerated dark space for 30 minutes to allow spurious scintillator activity to die away. The samples are then advanced one at a time to the counting station and the counts taken and printed out on paper tape. The sensitivity of the spectrometer for tritium activity depends on the counting time, the sample volume, and the background rate. For the secondary water samples, a counting time of 10 minutes will give a maximum sensitivity of about $1 \times 10^{-4} \mu\text{c}/\text{cm}^3 \text{H}_2\text{O}$ for tritium.

9.6.3 FISSION PRODUCT MONITORS

9.6.3.1 RRa-100, Fission Product Water Monitor. The fission product water monitor, RRa-100, consists of a Tracerlab MWP-1A unit which continuously samples the primary D_2O for fission products. The instrument isolates short-lived iodine activities from both the gross fission product and background activities in the primary coolant by ion exchange resin beds and gamma ray energy discrimination. Particulate matter is removed from the sample stream by a glass wool filter. Corrosion products and certain fission products are removed by a cation exchange column. The remaining fission products (anions) are collected in an anion column. The integrated activity in the anion column is measured by a Tracerlab MD-5B gamma scintillation detector assembly. The detector output is transmitted to a Tracerlab RM-20BS log ratemeter equipped with a spectrometer input circuit and to a DeVar multipoint recorder, both located in the control room. Ratemeter discriminator settings of 0.8 Mev to 1.3 Mev insure that the principal activities measured by the instrument are those of I^{135} , with contributions from I^{132} and I^{134} . The ratemeter output passes through a meter relay which actuates an alarm horn in the control room radiation equipment cubicle when the indicated activity exceeds a preset alarm level.

The D_2O which passes through RRa-100 is drawn from the head end of the D_2O purification system. Taps are provided to draw the sample stream from points either up or downstream of the filters FD-101A and B (see Figure 7.6). The D_2O from RRa-100 is discharged to the DA drain system and is collected in FA-101. The D_2O passing through the anion column, i.e., at the detector location, must not contain any appreciable N^{16} activity. A delay time of several minutes is, therefore, required between the primary system and the fission product monitor. This delay is easily achieved by passing the D_2O at a slow rate through a sufficient length of stainless steel tubing. The delay which is required for N^{16}

decay also delays the recognition of a fuel element failure, however, and for this reason the fission product gas monitor, RRa-105, is provided. The gas monitor operates in a relatively high background area and is less sensitive to small traces of fission products than the water monitor, but will respond more quickly to a significant fuel element rupture.

9.6.3.2 RRa-105, Fission Product Gas Monitor. The reactor cover gas radiation monitor, RRa-105, provides a means of detecting fuel element failure by continuously monitoring the gamma ray activity of the helium cover gas. The detecting device consists of a shielded Tracerlab MD-5B gamma scintillation detector positioned against a section of the helium circulation line 1-1/2 in. P113 (see Figure 7.5). The output of the detector is indicated on a Tracerlab RM-20B ratemeter and a DeVar multipoint recorder, both located in the control room. A meter relay on the ratemeter output activates both visual and audible control room alarms when the indicated activity exceeds a preset value.

9.6.4 STACK GAS AND PARTICULATE MONITORS. The radiation monitors RRa-305 and RRa-306 continuously measure the particulate and gas activity, respectively, of the exhaust air from the plant after it has passed through the building exhaust filter bank. The instruments are mounted in the filter room on the equipment level (see Figures 3.3 and 3.25). The air sample is drawn through an isokinetic sampling nozzle into both a Tracerlab MA-1A filter tape mechanism (RRa-305) and a MG-1A gas sampler (RRa-306). The sampling system flow is provided by a Miehe-Dexter 10 cfm sliding "Drivane" pump driven by a 1 hp electric motor.

The particulate matter in the exhaust sample is deposited on a moving filter tape and the activity on the tape is detected by a Tracerlab MD-1B beta-gamma detector. The detector, in conjunction with the tape transport mechanism, is capable of detecting a concentration of 3.0×10^{-12} $\mu\text{c}/\text{cm}^3$ of Sr^{90} or Y^{90} , or 8.0×10^{-12} $\mu\text{c}/\text{cm}^3$ of Co^{60} .

The filtered air is passed through a Tracerlab MG-1A gas sampler which provides a 0.1 ft^3 sampling volume enclosed by three inches of lead shielding. The detector is a Tracerlab MD-5B gamma scintillation detector which is capable of detecting a concentration of 1.7×10^{-6} $\mu\text{c}/\text{cm}^3$ of I^{131} in the air.

The output of each detector is indicated on a Tracerlab RM-20R log ratemeter and a multipoint recorder, all located on the control room radiation equipment panel. Meter relays on the ratemeter outputs actuate both visual and audible control room alarms when the measured activities exceed preset levels.

In addition to the monitors RRa-305 and RRa-306, a stack effluent gas sample is drawn through a charcoal filter for low-level iodine release detection. The charcoal filter unit and the associated pumping equipment are also located in the filter room. The iodine activity collected in the charcoal filter is counted periodically to determine the total iodine release from the HFBR complex. The activity counted is that of I-131 . The concentration of the effluent iodine activity in the charcoal filter gives a much greater sensitivity than is possible in the continuous flow gas monitor, RRa-306. Iodine-131 concentrations in the range of the AECM-0502 permissible concentration of 3×10^{-10} $\mu\text{c}/\text{cm}^3$ (see reference 11.2, Section 11) are detectable with the charcoal filter unit.

9.6.5 NUCLEAR INCIDENT ALARM SYSTEM. In the event of a release of radioactive material within the HFBR building, the nuclear incident alarm system automatically closes the building air intake lines, partially closes the exhaust line, closes the air conditioning unit condensate drain valves, and shuts down the building exhaust blowers. In addition, an alarm signal is transmitted to

the laboratory Security Office. A diagram of the system is shown in Figure 9.32.

The radiation detection equipment consists of two duplicate channels with detectors located on opposite sides of the balcony level. The detectors are Tracerlab Type TA-6 halogen-quenched G-M tubes with integrally mounted transistor ratemeter circuits and remotely actuated operational check source assemblies. The detectors have additional low level sources attached to them to provide an "up-scale" signal during normal low background conditions. The detector and ratemeter range is 0.1 to 100 r/hr.

The output of each channel is indicated on a Tracerlab TA-9 single channel indicator located in the control room. Each ratemeter controls an alarm unit containing high and low alarm contacts. The low alarm contacts monitor the failure of the channel to operate, as indicated by the loss of the low level source signal from the detector unit. The low alarm contacts of the two channels are connected in series to activate an annunciator point in the control room. The high alarm trip point will normally be set at 30 r/hr.

The high alarm contacts control the coils of the nuclear incident alarm relays K-114A and K-114B. The normally open contacts of these relays are arranged in pairs in the control circuits of the air conditioning unit condensate drain valves and the building containment 30 in. butterfly valves. Each of these circuits consists of solenoid-operated three way valves supplying 15 psig air to bellows type operators at the 30 in. butterfly valves and the condensate drain valves. The opening of the contacts in the alarm relays K-114A and K-114B de-energizes the solenoid valves and thus dumps the air from the valve operators and allows the butterfly valves and the condensate drain valves to close. The opening of the K-114A or K-114B relay contacts also gives an annunciator alarm in the control room and activates the Security Office alarm.

The building exhaust blowers are stopped by the action of relay K-115, with its coil in parallel with the solenoid control valve for the exhaust butterfly valve. A manual reset button for this relay is located in the blower room and prevents the automatic restart of the building exhaust blowers when the contacts on K-114A and K-114B close. This permits the inlet butterfly valves to be opened prior to the startup of a building exhaust fan.

The nuclear incident alarm relays K-114A and K-114B and the blower stop relay K-115 are Allen Bradley Type E220A, operating on a coil voltage of 125 VDC. The relays each contain four sets of normally open contacts. The contacts are rated at 10 amp for heavy duty service. The relay coils operate at .12 amp, 125 VDC. The relay operating time is approximately 50 milliseconds.

The solenoid valves are Type 8315B2-11, manufactured by the Automatic Switch Company. They are 1/4 in. size 3 way valves with a coil rating of 15 watts at 125 VDC.

The alarm units are Devar alarm modules with adjustable high and low level trip switches. The switch contacts are rated at 0.2 amp, 125 VDC and carry the relay coil currents of .12 amp. The switch contacts open to trip the incident alarm system.

9.6.6 BUILDING AIR ACTIVITY MONITORS. The particulate activity level of the building air is monitored by three fixed station units and one portable unit. Two fixed station units are located on the operations level, with sample points in the return ducts to the air conditioning units IB-301 and IB-302. One fixed station unit is located on the equipment level, with the sample point in

the canal area. The portable unit is normally located on the experimental level, but may be moved wherever needed.

Each air monitor consists of a vacuum pump, a moving filter tape deck, a gamma detector, a ratemeter circuit, a recorder, and an alarm unit. The recorders and alarm units for the operations level fixed stations are located in the health physics office on that level. The canal area station recorder and alarm are located with the other components of the unit.

9.6.7 TRITIUM AIR MONITORS. Tritium activity in the air around the pile top, in the reactor pit, in the equipment level shielded cells, and around the D₂O purification system is monitored by four stations. Each station has a 16 liter Kanne chamber as the detecting element. Air is drawn through the chamber from the sampling points by a vacuum pump. The chamber output is amplified in a Keithley electrometer and is fed to a recorder.

9.6.8 RADIATION MONITORING SYSTEMS DURING INITIAL LOADING AND START-UP. All radiation monitoring systems will be fully operative during the initial loading of the reactor and for all start-up experiments.

9.7 CONTROL ROD DRIVE MECHANISMS

9.7.1 INTRODUCTION. The control rod system provides the means for reactivity control in the HFBR. A total of 16 neutron-absorbing blades are moved close to the active core to achieve the control function. The blades are separated into two groups; 8 main blades operate above and alongside the core, and 8 auxiliary blades operate below the core. The blade geometry is discussed in Sections 4.1 and 4.3 and details of the blade construction are given in Section 4.3. The reactivity characteristics of the control rod blades and the shutdown margin are discussed in Section 4.6. The several modes of operation of the rod system are listed in Section 9.1.2, and normal start-up, operation, and shutdown are discussed in Section 12.6. The operation of the automatic power control channel is treated in Section 9.3.4, and the rod control switching and relay circuits, as well as the scram and setback circuits, are discussed in Section 9.5. In this section descriptions of the actuating mechanisms and position indicator will be given, covering design, functional requirements, and testing.

The control rod mechanisms and position indicator were obtained from the Diamond Power Specialty Corp. of Lancaster, Ohio on a performance specification. Preliminary studies established the functional requirements for the drives and the spatial limitations in the reactor vessel and shield pit. The mechanism design was carried out by the Diamond Power Specialty Corp. Comprehensive testing of a prototype mechanism to prove the design was a requisite of the specification. Further testing of each drive produced for the reactor was also required.

Because of the critical nature of the drive mechanisms, great stress was placed upon simplicity of design using machine elements of proven reliability. The choice of materials was limited to those best suited to the environment, with restrictions placed on high cobalt alloys and untested materials. All drawings, calculations, materials, manufacturing processes, tests, and quality control procedures required the approval of The Lummus Company and were subject to review by the BNL staff.

9.7.2 FUNCTIONAL DESCRIPTION

9.7.2.1 Geometry. The basic geometry of a control rod drive, mounted on the reactor vessel, is shown in Figure 9.22. The control blades are attached to

the driving elements by couplings (see Section 4.3.7) above the transition plate. The drive train for a main rod is shown in the expanded view in Figure 9.23. The portion of the mechanism inside the reactor vessel is shown in Figure 9.24. An enlarged view of the portions of the mechanisms and blades near the transition plate is given in Figure 9.25. Figure 9.26 shows details of the shaft seal. Figure 9.27 is a plan view of the reactor pit, showing the layout and numbering scheme for the drive mechanisms. Figure 9.28 is a schematic drawing of the position indicator system.

Although each of the 16 blades is individually driven, it has been convenient to group one main and one auxiliary drive into a single assembly to obtain maximum rigidity in a minimum space envelope. This grouping gives rise to the nomenclature "M-1, A-1, M-2, A-2", etc., through "M-8, A-8", where "M" represents main drives, "A" auxiliary drives, and the numerals designate position, advancing clockwise in sequence from grid north. This nomenclature is shown on Figure 9.27. It has been impossible to construct 8 identical (and thus interchangeable) units because of interference problems in the crowded neck of the vessel. Instead, the odd numbered units are "right hand" drives and the even are "left handed". Internal differences between left and right hand drives are discussed in Section 9.7.2.7. The main control rod blades are completely interchangeable, as are the auxiliary blades.

Each of the 8 drive assemblies has 2 major subassemblies. The first subassembly is the long dual tube support structure containing racks, pinions, and blade couplings. This assembly is mounted within the reactor vessel. The assembly is bolted to the vessel at the upper end and is guided by pilots on the flow shroud at the midpoint and bottom. This mounting arrangement can be seen in Figures 9.22 and 9.24. The other subassembly consists of motors, reducers, an electromagnetic clutch, limit switches, and position transmitting equipment and is located outside the reactor vessel. The two subassemblies are mechanically connected through seal cartridges.

The internal subassembly, including the rack and pinion housing, was not allowed to cover any portion of the core, to permit free vertical access to fuel elements during reloading. This requirement led to offset connections between the blades and drives. The offset distance was reduced as much as possible by mounting the main and auxiliary support tubes close together. The offsets from the centroids of the blades to the centerlines of the support tubes and followers are approximately 1-11/16 in. for the main rods and 3-7/8 in. for the auxiliary rods.

The external drive components are mounted on 8 frame assemblies which bolt to the reactor vessel near the shaft penetrations at elevations 131 ft-4 in. and 132 ft-1-1/2 in. Each drive and frame assembly (called a power unit) contains one complete auxiliary and one complete main drive unit. Both units connect through seal cartridges to driving pinions within the vessel. The centerline of the auxiliary drive is located 9-1/2 in. above the main drive centerline. Bolt- ing the power unit to the vessel insures accurate alignment of the internal and external drive shafts.

Although most individual components of the power units are interchangeable, there are 4 types of power units assemblies; 3 "right hand" drives (numbers 3, 5, and 7), 3 "left hand" drives (numbers 4, 6, and 8), 1 right hand drive modified for automatic control (number 1), and drive number 2 with a special frame and extended shaft. The latter drive is designed to clear the fuel discharge chute, which penetrates the service floor approximately 5 feet from the center of the vessel and rises at an angle of about 16° from the vertical. The chute would interfere with a standard power unit.

9.7.2.2 Environment. The support tube structures located within the reactor vessel are designed for the following conditions:

Immersion in high purity heavy water.
 Helium atmosphere above 133 ft-6 in. elevation.
 Temperature: 150°F
 Pressure: 275 psi
 D₂O flow: 16,600 gpm
 Thermal cycle: 50° to 150°F
 Gamma Dose Rate:
 At lower end (coupling): 10^8 r/hr (0.3 watts/gm)
 At rack and pinion housing: 10^3 r/hr
 Neutron Flux:
 At lower end (coupling): 10^{13} n/cm²-sec
 At rack and pinion housing: 10^7 n/cm²-sec

The power units and seal assemblies outside the vessel are designed for the following conditions:

Atmosphere: stagnant air
 Temperature: 150°F
 Relative Humidity: 85%
 Gamma Dose Rate: 100 r/hr
 Neutron Flux: 10^6 n/cm²-sec

9.7.2.3 Normal Operation. The reactor is brought to power by withdrawing main and auxiliary rods in "ganged" motion, one set at a time. Through most of the operating cycle the two sets of rods are symmetrically located above and below the core. The power level is maintained at a constant value by automatic control of one auxiliary rod. The automatic control rod is rod A-1, actuated by drive A-1. During normal operation the maximum strokes for the main and auxiliary rods are 30 and 15 inches, respectively. Both rods move at a speed of approximately 4.3 in./min in either direction.

The rods are moved by energizing the electromagnetic clutches in the main rod drives and turning on the electric motors in both main and auxiliary drives. As pointed out in earlier sections, prescribed conditions of pressure, flow, and liquid level must be reached in various reactor systems before the main rod clutches can be energized.

The switching circuits for main and auxiliary rod motors are shown in Figure 9.19. Both main and auxiliary rod motors are permanent magnet synchronous units operating on 120 volt, 60 cycle, single phase power. Each motor has two electrically identical windings. To actuate the motor, one winding is connected by the rod control switches directly across the 120 VAC line. The other winding is connected by the switches to the line in series with an RC phase shifting network. To reverse the motor the two windings are interchanged by the control switches with respect to the power line and phase shifting network.

The main drive train consists of a 72 rpm motor coupled to a gear reducer whose output shaft rotates at .605 rpm. As shown in Figure 9.23, an electromagnetic clutch is keyed to the reducer and, when energized, couples the motor to a spring arbor and to a drive shaft. The drive shaft terminates in a seal cartridge mounted in the wall of the reactor vessel. As the shaft and arbor turn, a power spring is wound up to store some of the energy that is required in the event of a scram. On the reactor side of the clutch, a small, low inertia gear train is used for position indication, scram timing, and to actuate a cam for rod position limit switches.

The seal cartridge contains a hydraulically balanced spring-loaded face-type seal which is the boundary between the pressurized primary system and the atmosphere. The drive shaft of the external drive subassembly is joined to a pinion shaft on the internal subassembly by a splined coupling (see Figure 9.26). Rotary motion of the pinion shaft is translated to vertical movement of the rod through a rack and pinion. Upward rack motion (rod withdrawal) results in compression of a helical spring located within the tube support housing, thus adding to the stored energy of the system.

The main rack assembly is a combination of several parts; an upper rack section, a lower follower which is coupled to the blade, and an internally threaded piston connecting the rack and follower (see Figure 9.24). The piston, in concert with a snubber at the bottom of the support tube, decelerates a scrambled rod. A collar at the top of the rack section guides the rack within the tubular housing. The rack can be driven upward until the collar contacts a projection near the upper limit of its travel. The projection is a hard stop and is used as a fixed reference from which blade travel measurements are made.

The auxiliary drive train has the motor and reducer tied directly to the drive shaft without a clutch or power spring. The rack and pinion motion is the same as in the main drive but there is no piston and snubber in the auxiliary guide housing. An extension of the auxiliary rack follower terminates in a coupling which attaches to the auxiliary blade. Downward motion of the rack (corresponding to blade withdrawal) compresses a helical spring mounted outside the support tube. This spring serves two purposes; it reduces backlash in the system and associated errors in position indication, and drives the auxiliary blade to the fully inserted position in the event of a break in the drive line.

Table 9.7-1 gives the rod speeds, strokes, and tolerance limits for the main and auxiliary drives.

Table 9.7-1 Control Rod Speeds, Strokes, and Tolerance Limits

	<u>Main Drive</u>	<u>Auxiliary Drive</u>
Stroke:		
Normal operation	30 in.	15 in.
Blade replacement	36 in.	15 in.
Motor speed	72 rpm	72 rpm
Reducer speed, output shaft	.605 rpm	.329 rpm
Blade speed, both directions	4.35 in./min	4.28 in./min
Max. deviation in speed between drives	0.10 in./min	0.10 in./min
Max. lateral movement of blade coupling during stroke	±0.015 in.	±0.015 in.
Max. coasting, both directions	0.010 in.	0.010 in.
Drift, creep	0	0
Max. deviation in elev. of blade couplings, with drives mounted in vessel	±0.010 in.	±0.010 in.
Accumulated tolerance between coupling & tip of blade	±0.025 in.	±0.025 in.
Accumulated tolerances within position transmitter, receiver, and linkage	±0.065 in.	±0.065 in.

9.7.2.4 Reactor Scram. A scram action can be initiated by any one of a number of potentially unsafe conditions, as discussed in Section 9.3. The scram action interrupts current to the electromagnetic clutch, which, on opening separates the motor, reducer, and stationary portion of the clutch from the balance of the drive. The action of both the power and helical springs thrust the main blades downward to their fully inserted positions. At the same time the auxiliary blades are inserted at normal speed to be in position for the next reactor startup.

It was necessary to set dynamic requirements for the scram insertion at an early date in the reactor design. As a result of preliminary studies (9.6), an initial travel of 10.6 inches from the fully withdrawn position within 180 milliseconds of the occurrence of an unsafe condition was set as the scram requirement. However, the actual time allotted for motion was 145 milliseconds, since 15 and 20 milliseconds, respectively, were assigned to instrument response and clutch current decay. Using standard kinetic formulae, it was shown that the main blades required an average acceleration of $1008 \text{ in. sec}^{-2}$ and were moving at a velocity of 146 in./sec at the end of 145 milliseconds. A time-displacement curve was drawn having the above acceleration for the initial 10.6 inches of travel and continuing for approximately 5 inches at a constant velocity of 146 in./sec. Although a scram could occur anywhere along the stroke, the fully withdrawn position is the one from which it is most difficult to insert shutdown reactivity rapidly (see Section 4.6). The specifications limited the actual motion of a scrambled blade to a path below the curve described above.

As noted in Section 9.7.6, the prototype drive mechanism exceeded the specification requirements for the scram action, as did each production drive unit for the reactor. The actual scram motion of the drives is $12\text{-}7/8$ in. from the fully withdrawn position in 175 msec, including the release time of the clutches (see Figure 9.31). With the nuclear safety channel and safety system response times of 1 and 5 msec, the overall control system is capable of inserting 4.4 %k in control rod reactivity effect in 181 msec in response to a fast nuclear transient.

The internal scram spring and external power spring are sized to deliver a minimum force of 420 pounds to the blade at the fully withdrawn position. This force is required to accelerate the vertically moving components, rotate the horizontal components, and overcome the friction in the shaft seal. The acceleration of the vertical components, in combination with the vertical forces resisting this acceleration, requires almost 75 percent of the total stored energy. The resisting forces include the buoyancy effect, the hydraulic and frictional resistance to scram motion, and the inertia of translating parts. No credit was taken in the calculations for forces from coolant flow or core pressure drop, which act to assist the scram motion.

Approximately three-tenths of the required energy is supplied by the internal scram spring. There are several advantages in introducing energy into the drive train at a point near the blade. These include lower buckling stresses on the rack and a minimum number of intermediate members between the driving force and the driven element. Every effort consistent with good design practice was made to maximize the force available from the internal spring. Stringent space limitations prevented the use of a more powerful internal spring. The internal spring alone will move the blade 10 in. from the fully withdrawn position in 360 milliseconds and 14 in. in 500 milliseconds.

The balance of the scram force is supplied by a power spring located in a housing outside of the reactor vessel. Figure 9.29 shows the contribution of each spring to the total force as a function of blade position and shaft

rotation. An important advantage of the dual spring system is that the pre-stress levels are adjustable on the power spring, in increments of 45° from 0° to 135° from a point outside the vessel. Pre-stressing flattens the force-displacement curve, and permits a more constant energy gradient. Furthermore, the pre-stressing permits adjustments to be made in the available scram energy, so that scram times can be made uniform on all drives.

The energy of the moving blade and drive line components must be dissipated in a manner which will permit full blade insertion in as short a time as practicable without imposing excessive forces on the drive elements. For the first 10 inches of scram insertion from the fully retracted position, water under the descending piston can escape through ports in the support tube. The ports do not extend below this point and the heavy water must exit through the annuli between the piston and support tube and between the snubber and follower shaft. This dashpot arrangement retards the blade until a final stop is made when the upper rack collar strikes a stack of belleville washers in the pinion housing. The leakage paths around the piston and snubber are of the pressure-breakdown type. The piston has circumferential lands and grooves on the outer surface which cause hydraulic entrance and exit frictional energy losses as the water is forced between the piston and the support tube. The snubber has the same land and groove arrangement on the inside, in the hole through which the rack follower passes.

As the piston and drive train move downward in a scram, the frictional force opposing the motion peaks sharply as the piston enters the dashpot region. The magnitude of the force peak is quite sensitive to the piston and snubber clearances. A series of calculations (9.7, 9.8) were made to determine the correct clearances, and the results of these are shown in Figure 9.30. Note that Case II, giving a peak force of 1912 lb, was adopted for the mechanism design.

The snubber force peak is an impact load on the drive train during the scram. The components of the train below the piston are placed in tension, as is the lower main support tube, and the peak force leads to peak stresses in these components of only 1160 psi in tension, as shown in the prototype test (9.9; also see Table 9.7-8). The critical drive components, so far as the snubber impact force is concerned, are the rack and the rack and pinion gear teeth. The rack loading is a buckling force of 1912 lbs. The unsupported rack length is 46 in., with the lower end essentially fixed and the upper end laterally restrained. The critical buckling load is calculated to be 3.5 times the applied load (9.7). The rack and pinion gear teeth are designed to withstand the combined forces of the snubber load and the power spring, which are additive in the scram. The many scram tests on the prototype mechanism and on the production mechanisms bear out the adequacy of the design.

More than half of the total scram energy of 840 ft-lb is expended in the snubber. The remaining energy is dissipated as friction in the drive line components and in the dashpot effect of the main rack collar operating within the upper support tube. A small part of the scram energy is expended in friction in the belleville washer stack.

The results of scram time tests are shown in Figure 9.31. Rod position vs. time curves are given for scrams originating at various main rod positions up to and including the fully withdrawn position.

The case of a scram with no water in the snubber region (dry scram) was examined to make sure that no damage to the reactor vessel or core structure would result from such an event. It must be pointed out that nuclear safety was

not involved in this study since a dry scram could only occur when the reactor was not in operation. When the reactor is shut down it is possible to withdraw blades by use of a hand drive. To operate the hand drive it is necessary to lock out the power spring from the drive line. Thus a dry scram, which can result from the stripping of rack or pinion teeth or the fracture of a component, will release only the energy of the internal coil spring. Without water in the vessel the moving drive line elements are not decelerated in the snubber and the upper rack collar strikes the belleville washer stack at terminal velocity. The washer stack collapses downward a total distance of 0.75 in., absorbing over 90% of the scram energy in the process. The remaining energy is dissipated in the rack and follower without excessive stresses in any component (9.7). The .75 in. downward overtravel of the drive line does not result in the blade coupling striking the transition plate.

A more severe dry scram case is that in which the rack is fractured, so that the cushioning action of the belleville washers is not available. In this case the piston strikes the snubber section of the main support tube, and the scram energy is absorbed by the tube structure. The stresses in the various members in this case are above the yield point but do not exceed the ultimate strength of the material (9.7). Some deformation of the piston, snubber, or housing tube might occur, and the drive would have to be replaced. No damage to the reactor vessel or core structure would result. The force transmitted to the reactor vessel at the top flange for this maximum dry scram case would not exceed 45,500 lb per drive (9.7).

9.7.2.5 Automatic Control. Automatic control of the reactor to maintain a constant power level is carried out by devices in nuclear instrument channel #6 (see Section 9.3) which position auxiliary rod number 1 (A-1). The drive unit for rod A-1 includes several components of the automatic control system and is, therefore, not identical to the other drive units.

The components in the A-1 drive associated with automatic control are geared to the unit drive shaft. They are a slide wire potentiometer and a differential transformer. The potentiometer provides a DC signal to the rod position controller unit at the console so that the controller can be adjusted to match the position of rod A-1 prior to automatic operation. A position signal from the controller energizes an interlock that limits total reactivity in the automatic mode to less than 1 dollar. An alarm circuit, operating on signals from the controller, sounds an alarm when the position of rod A-1 deviates by more than one inch from a preset point which represents the average auxiliary rod "crown" position. A manual adjustment of the crown position is then made, and the alarm reset.

The differential transformer provides an AC electrical signal proportional to rod position for feedback to the rod positioner unit in nuclear channel #6. The rod positioner unit compares the actual rod A-1 position, as given by the differential transformer signal, to the required rod position to maintain constant power, as given by the rod position controller signal. The positioner output is an on-off signal to the motor of drive A-1 which moves the rod to the required position.

9.7.2.6 Hand Drive. Some means of operating the rod mechanisms during shutdown periods is necessary to permit adjustment of limit switches, resetting of the position indicator, and changing of the blades. The drive motors and clutches are locked down during shutdown periods so that normal, electrically powered operation is not possible. Further, the main rod clutches cannot be energized unless all the scram instruments are at normal operating points, and this requires primary system pressurization and full flow. Hand drives for each

rod are, therefore, built into the power units.

With the motor immobilized and the clutch decoupled, the main drive is actuated by means of a worm and worm gear located on the reactor side of the clutch. The worm is mounted in a fixed housing and is driven by a ratchet-wrench tool through a set of spur gears. The worm gear is bolted to the power spring housing. The hand drive rotates the spring housing and arbor as a unit, and thus rotates the drive shaft. It takes 19.8 revolutions of the hand drive tool for each inch of blade movement.

During normal operation the worm gear and power spring housing are bolted to the fixed worm housing by bolts and do not rotate. For manual operation it is necessary to remove the bolts and insert keys between the power spring housing and the clutch cup. The keys maintain the power spring in its preloaded position. Keys are removed and bolts replaced when returning to the normal mode of operation.

Should the drive be left with the hand drive connected, i.e., with the keys in place and the bolts not replaced, and an attempt made to operate the drive motor, the motor will stall. No damage will result, since the motors are a type which can be stalled indefinitely without overheating. The rod cannot be motor-driven with the hand drive keys connecting the power spring housing and clutch cup.

If the keys are removed, but the power spring housing is not bolted to the worm housing, the rod can be moved by the motor in the normal way. In this case the power spring housing is held stationary by the engagement of the worm and worm gear, rather than by the bolts. The worm and worm gear are quite strong enough to withstand the torque of the power spring. Some drifting of the rod from a given raised position might occur if system vibration caused the worm to rotate slightly under the load.

If the hand drive is operated without the keys which lock the power spring housing and spring arbor together through the clutch cup there is the possibility of unwinding the power spring and breaking it. This difficulty can be avoided by coupling the hand drive only when the rod is fully inserted. In this rod position the rod can only be raised, which winds the power spring in the correct direction. Proper use of the keys, of course, avoids the problem.

The auxiliary hand drive is a simpler unit, since there is no power spring involved. For normal motor-driven operation the worm gear is bolted to a fixed worm housing and torque is transmitted directly to the drive shaft through a splined adapter and coupling. To operate the hand drive, the bolts are removed to free the worm gear and set screws are used to connect the worm gear and splined adapter. The reducer coupling is then disengaged from the splined adapter. If, by error, the set screws are not removed prior to power driving, the motor will stall since the worm gear locks against the worm.

9.7.2.7 Right and Left Handed Units. The geometry of the 6 x 7 rectangular array of fuel elements and the congested area above the core forced the adoption of "right" and "left hand" drive assemblies. Both the rack and pinion assembly within the vessel and the external drive unit are affected by this requirement. The odd-numbered drives have been designated "right hand" and the even-numbered drives "left hand". Differences in the two sets of mechanisms are treated in the following paragraphs.

Within the vessel the drives differ in three places, at the dowel pin plate assembly, the lateral guide plates, and the rack and pinion housing. Dowel pin

plates have two projections or ears, one longer than the other. Because of the drive spacing the longer ears are subdivided into two lengths. The dowel pin plates are interchangeable for drives 1 and 5, 2 and 6, 3 and 7, and 4 and 8. The rack and pinion housings differ in that the pinions are to the right of the racks in the right hand units (facing the core), and to the left in the left hand units. The lateral guide plates, like the dowel pin plates, have two projections of different length, the right projection is longer for right hand units and the reverse for left hand units.

Outside the vessel, differences in operation due to left and right hand units are as follows:

	<u>Right Hand</u> (1, 3, 5, 7)	<u>Left Hand</u> (2, 4, 6, 8)
Direction of output shaft of reducer for withdrawal of main blade	clockwise	counterclockwise
Direction of output shaft of reducer for withdrawal of auxiliary blade	counterclockwise	clockwise
Rotation of hand drive for withdrawal of main blade	clockwise	clockwise
Rotation of hand drive for withdrawal of auxiliary drive	counterclockwise	counterclockwise

The clockwise and counterclockwise senses of rotation are taken from a position behind the drive, facing the core, for horizontal shafts, and from above for vertical shafts.

In order to keep the rotation of the hand drives consistent, i.e., clockwise rotation for both left and right hand main drives to raise a blade, right hand worm gearing was used in right hand drives and left hand gearing in left hand units. Left and right main drive shafts rotate in opposite directions while the blades travel in the same direction. The power spring and housing are so arranged that clockwise rotation of right hand units and counterclockwise rotation of left hand units wind the power spring as the rod is withdrawn.

Although components are clearly identified, it is not impossible for errors to occur in the assembly of the drive units which might result in awkward combinations of right and left hand parts in the same drive line. Should a right hand withdrawal assembly be used to drive a left hand internal assembly, a blade withdrawal signal would result in a torque in the direction of insertion. Since the drive would be against the lower hard stop in this case the motor would stall. If an opposite hand power spring were inadvertently installed, withdrawal would unwind the spring and result in spring failure. In a similar way an incorrectly wired motor could result in unwinding and overstressing the power spring. In all of these cases the operating check and scram time test made on each new drive installation before startup would reveal the difficulty.

9.7.3 DESCRIPTION OF EXTERNAL COMPONENTS

9.7.3.1 Motors and Gear Reducers. The main and auxiliary drives use the

same types of permanent magnet synchronous AC motors and the same type of single worm helical gear reducers. The motors were chosen because of their high torque, slow speed, and fast start, stop, and reversal characteristics which insure a high degree of synchronism between units. No overload protection is necessary since the motors can be stalled continuously without damage. Motor and gear reducer ratings are given in Table 9.7-2.

Table 9.7-2 Control Rod Drive Motor and Gear Reducer Ratings

<u>Motor</u>	<u>Main Drive</u>	<u>Auxiliary Drive</u>
Manufacturer	Superior Electric Co.	Superior Electric Co.
Model	Slo-Syn Synchronous Type SS400	Slo-Syn Synchronous Type SS250
Input power	120 V, 60 cycle, 1 phase	120 V, 60 cycle, 1 phase
Max. current	0.6 amp	0.6 amp
Output speed	72 rpm	72 rpm
Output torque	400 inch-ounces	250 inch-ounces
Insulation	Class H	Class H
<u>Reducer</u>		
Manufacturer	Morse Chain Co.	Morse Chain Co.
Model	No. 25W-HB	No. 25W-HB
Reduction ratio	119:1	219:1
Output speed	0.605 rpm	0.329 rpm
Output torque (at full speed)	970 inch-pounds	660 inch-pounds

9.7.3.2 Main Drive Clutches. The main drive clutch characteristics are given in Table 9.7-3. The clutch is an electromagnetic device located in the drive train between the gear reducer and the power spring assembly. There is no clutch in the auxiliary drive unit.

When the clutch coil is energized the clutch cup and a set of discs are locked to the main clutch housing. The cup and discs are rapidly released from the housing when the coil current is interrupted. Since the cup and disc are of light weight they contribute little to the total inertia of the drive during a scram.

9.7.3.3 Power Spring Assembly. The power spring consists of a strip of tempered steel wound on an arbor and confined in a drum housing. The power spring stores the energy used in winding it and delivers the energy through rotation of the arbor when the clutch decouples the arbor from the motor and gear reducer. Spring data is given in Table 9.7-4.

The clutch cup shaft is keyed to the arbor. The arbor is slotted to receive the inner end of the power spring. The outer end of the spring is pinned to the spring housing, which is stationary. The spring arbor also contains an internal spline into which the drive shaft is fitted. Thus, the rotation of the

motor is transferred through the gear reducer, clutch, and arbor to the drive shaft. Pre-winding of the spring for about 45° is essential to improve its load-deflection characteristics and to permit rod insertion overtravel without reverse winding of the spring. If the spring is forced to unwind upon itself failure can occur. Pre-winding is achieved by rotating the spring housing with the hand drive while the arbor is held from turning because of its direct connection to the fully inserted blade. At the desired spring preloading, bolts are inserted to lock the spring housing in a stationary position.

Table 9.7-3 Main Drive Clutch Characteristics

Manufacturer	Carlyle Johnson Machine Co.
Model	"Maxitorq", Series 9000, No. 475
Coil rating	100 VDC, 0.5 amp, 50 watt
Coil resistance	150 ohms at 68°F
Dynamic torque rating	75 ft-lb
Holding torque rating	150 ft-lb
Coil current for 75 ft-lb torque	0.325 amp
Normal operating current (coil hot)	0.360 amp
Coil voltage supply	67.5 VDC
Release time	12 milliseconds

Table 9.7-4 Power Spring Data

Material	SAE 1074 spring steel
Dimensions	.072 in. thick 2 in. wide 207 in. long 1.800 in. arbor diameter 6.750 in. housing inside diameter
No. turns for 30 inch travel	4.18, or 1500° of rotation
No. turns to wind solid	6.54, or 2350° of rotation
Torque	400 inch-lbs, solid wound 360 inch-lbs, 4.18 turns (fully withdrawn rod)
Stress	190,000 psi, solid wound 170,000 psi, 4.18 turns (fully withdrawn rod)
Force-displacement curve	Figure 9.29

9.7.3.4 Drive Shaft and Seal Assembly. The drive shaft and seal cartridge unit are pre-assembled to assure an optimum shaft-to-seal fit. Except for the length of the drive shaft, all 16 assemblies are identical, and use the John Crane Type 9BT hydraulically balanced face seal shown in Figure 9.26. The seal contains a preloaded polyethylene wedge ring which prevents leakage between the shaft and a carbon washer. The washer has a precision lapped face which mates against a similarly machined surface on a stationary seat. The seal is similar to that used for pump shaft seals (see Section 5.3.2). The low friction construction of the seal offers minimum resistance to a scram.

The seal operates within a flanged cartridge which is accurately fitted into an opening in the reactor vessel. Bolts through the cartridge flange hold the unit in place. The cartridge is sealed to the reactor vessel by two O-rings. The cartridge is fitted with an inboard Graphitar bearing and an out-board ball bearing to support the drive shaft; the former operates in heavy water, the latter in air. The seal cartridge material is 304 stainless steel.

An O-ring is provided as backup for the mechanical seal. The annulus between seals has two vent holes for collecting leakage, one for gas and another for D_2O . The gas effluent is piped to the main off-gas manifold, EV101, and the liquid drain, DA109, returns the D_2O to the FA-101 storage tank (see Figure 3.25).

The drive shaft connects the internal control rod assembly to external driving elements through the seal cartridge. A male spline at the external end of the drive shaft engages the power spring arbor of the main drive, or the splined adaptor of the auxiliary drive. At the internal end of the drive shaft, a female spline receives the end of a spherically splined shaft which connects to the pinion gear. The drive shaft material is 440C stainless steel, as is the spring arbor.

9.7.3.5 Position Transmitter and Limit Switch Assembly. Both the main and auxiliary drives have individual position transmitter and limit switch assemblies geared to the respective drive shafts. The main transmitter assembly consists of a potentiometer geared to the drive shaft and a transmitter geared to the potentiometer shaft. The potentiometer is a 10,000 ohm Model 9303 Helipot. The transmitter is a Daystrom Type 11CT6 servo generator, operating on 6.3 volt, 60 cycle power. Spring-loaded split gears are used in the gear train to minimize backlash error.

The drive shaft turns 4.178 revolutions for 30 inches of blade movement. The rotation of the potentiometer shaft, geared down by a ratio of 42:75, is 2.34 revolutions of the rated 3 revolution capacity. The potentiometer is used to measure the main drive scram time. A battery is connected across the potentiometer winding and the voltage at the variable tap, which is proportional to rod position, is displayed on a persistent-screen oscilloscope or on a fast recorder. The voltage vs. time trace for the scram can be compared to previous traces for the same drive and to standard scram time-displacement curves.

The position transmitter is geared to the potentiometer shaft in the ratio 28:150 and makes 0.437 revolutions for the full 30 in. rod stroke. The transmitter supplies rod position signals to the control room position indicator panel, as described in Section 9.7.5. Figure 9.28 is a schematic drawing of the position indicator system.

A cam assembly is mounted on the transmitter shaft. Two cam lobes can be independently and precisely adjusted to actuate the "full-out" and "full-in" limit switches. The switches are double pole Microswitches, with one set of

contacts on each switch interrupting the motor current and the other set actuating full-in and full-out pilot lights on the position indicator panel.

The auxiliary drive position transmitter and limit switch assembly is similar to the main drive unit but has no potentiometer. The transmitter is geared to the auxiliary drive shaft with spring-loaded split gears to minimize backlash errors. The transmitter shaft turns 4.99 revolutions to 1.15 revolutions of the drive shaft for the full 15 inch stroke of the auxiliary rods. The transmitter is a Bendix Corp. Model 148-7278-1 torque transmitter (see Figure 9.28). The transmitter supplies rod position signals to the position signals to the position indicator panel in the control room, as described in Section 9.7.5.

The auxiliary drive limit switch assembly is driven from the drive shaft by a separate set of gears. Cams actuate two double pole Microswitches. One switch is the "full-in" limit switch and the other is the "full-out" limit switch. One set of contacts on each switch interrupts the auxiliary motor current and the other set actuates full-in and full-out pilot lights on the position indicator panel. The cam shaft is geared to turn 0.46 revolutions for the 1.15 revolutions of the auxiliary drive shaft corresponding to the full 15 inch auxiliary rod stroke.

The auxiliary drive A-1 is equipped for automatic control. As noted in Section 9.7.2.5, two components set this drive apart from the others. The first is a potentiometer which is geared to the transmitter shaft. For the full 15 inch blade stroke the potentiometer rotates 8.73 revolutions of its rated 10 turn capacity. The potentiometer is a 250 ohm Model 7603 Helipot. The other component is a differential transformer. A cam geared to the A-1 drive shaft changes rotary motion of the shaft to linear motion at the transformer. With the drive shaft rotating 1.15 revolutions for the 15 inch stroke and a gear tooth ratio between drive shaft and cam shaft of 48:65, the cam shaft rotates 0.851 revolutions for the full stroke. A 1/2 inch rise on the cam results in the movement of a spring-loaded arm which actuates the transformer armature over a range of 0.060 to 0.125 inches. The transformer AC output signal is proportional to the armature position. The cam profile is cut to give a linear relation between armature position and the A-1 rod position.

9.7.3.6 External Drive Assembly Mounting Frame. Each external power unit, consisting of both main and auxiliary drive trains, is built into a mounting frame which is bolted to the reactor vessel. The frame is constructed of 2-1/2 in. ASTM A-7 structural angles. The frame is approximately 11 in. wide, 24 in. high, and 31 in. long. The frame for No. 2 drive is 48 in. long, to accommodate the extended drive shaft of that unit.

The frame is attached to the reactor vessel by 3/4 in. bolts through four lugs welded to the vessel. The lugs are 6061-T6 aluminum alloy, with stainless steel bushings in the bolt holes. The set of four lugs is grouped around the two seal cartridge penetrations and flanges for each drive assembly on the reactor vessel.

The main drive electric motor is screwed to a plate which is bolted to the mounting frame. The main drive reducer, clutch, power spring and hand drive are pre-assembled on a base plate which is then bolted to the angle frame. Four horizontal screws on the frame bear on the corners of the base plate for lateral adjustment of the plate with respect to the drive shaft. The main rod position transmitter and limit switch sub-assemblies are fixed to an aluminum plate having slotted holes. These slots make possible proper alignment between the small gear on the drive shaft and its mating gear on the position transmitter prior to bolting the aluminum plate to the frame.

The auxiliary gear reducer and hand drive share a common base plate which is bolted to the frame above the main drive components. Lateral adjustment of this plate is made with jack screws similar to those used on the main drive. The auxiliary motor is mounted on a bracket similar to that of the main. The limit switch assembly is mounted upon a small plate which, in turn, is screwed to the auxiliary position transmitter base plate. The latter plate is fastened through slotted holes to the support frame. By this arrangement, independent adjustments may be made of the positions of the gear on the cam shaft (part of the limit switch assembly) and the transmitter gear, with respect to their mating gears on the drive shaft. The differential transformer and cam assembly is screwed to a bracket which is then bolted to the frame of the A-1 drive. Elongated holes in the bracket and base of the transformer permit a proper mesh of the cam shaft gear with its driver on the auxiliary drive shaft.

Maintenance of external drive components is made easier by the use of sub-assemblies on independent base plates. It is not necessary to remove the entire mounting frame or disturb the seal to replace or repair a component of the external assembly.

9.7.4 DESCRIPTION OF INTERNAL COMPONENTS

9.7.4.1 Rack and Pinion Housing Assembly. Both main and auxiliary racks pass through a common housing where they are engaged by the respective driving pinions (see Figure 9.24). Each pinion is mounted on a pinion shaft which rotates in Graphitar bearings. Each pinion shaft has a spherical splined connection to a second shaft which engages the external drive shaft at the seal cartridge. The racks move vertically through long Graphitar bearings which provide thrust support at the pinions. There are 4 "right hand" and 4 "left hand" housings; the difference between types depends upon whether the pinion drives from the right or left of the rack. The housings are machined from type 304 stainless steel forgings and weigh 80 pounds each.

There are side plates riveted to each pinion which contact flat surfaces on the rack outside of the rack teeth. This arrangement prevents rotation of the rack. The pitch diameter of the main pinion is 2.286 in., and of the auxiliary 4.143 in. The pinion is spline-mounted to the pinion shaft, to permit a small amount of "float" which reduces shock loadings and problems of misalignment. Of the two gears the main pinion is more highly stressed. It is designed to withstand a tangential tooth load of 2000 pounds which occurs during a scram from the "full-out" position. In the scram, the impulse loads due to snubber action are added to the force of the power spring. The tooth form adopted to meet the scram loading was a 7 diametral pitch, 20° stub involute tooth profile with a .786 in. face width. The same tooth form is used on the auxiliary pinion and on both racks. The pinion and sideplate material is 17-4PH stainless steel which is drawn at 1100°F as the final stage in the heat treatment. Pinion gears of type 304 stainless steel with 440C sideplates were tried in the first prototype tests and were found unsatisfactory from a corrosion standpoint. An alternate material for the pinions and sideplates is Inconel X, and a set of gears and shafts of this material has been obtained for testing.

The purpose of the two-shaft arrangement connecting the pinion to the external drive shaft is to provide a self-aligning coupling between the pinion and the external shaft. The outer shaft has 1 in. pitch diameter male spherical splines at both ends which permit up to 2 degrees of shaft misalignment. This is equivalent to a displacement of the support tube assembly with respect to the control rod vessel penetrations of 0.281 in. The main and auxiliary pinion shafts and splined shafts are subject to only moderate wear and stress. The shaft material is 440C stainless steel. The shafts are nickel plated and flash

chrome plated. The female splines are only nickel plated, to match the chromed male splines.

A stack of 25 belleville washers is sandwiched between flat washers and fits within a counterbored hole in the housing just above the main pinion. In the fully inserted position of the main rod the main rack collar rests upon this stack of washers, and in a scram the washers absorb the residual kinetic energy of the drive train at the end of the stroke. The belleville washers were selected to absorb the shock loads because of their large load capacity per unit volume, small deflection, and low closed height. The washers are stacked in series, that is, adjacent washers are reversed. The washer material is 17-7PH stainless steel. The impact load on the stack for a normal scram is 890 pounds, with a corresponding total deflection of 0.060 to 0.065 in. The decrease in height of the washer stack from the normal unloaded condition to the solid stack is 0.75 in.

There are 6 Graphite bearings mounted within the housing; 2 each for the main and auxiliary pinion shafts and 1 each for the main and auxiliary racks. The pinion shaft bearings are grade 14 Graphitar clad in type 304 stainless steel shells. These bearings are a slip fit in the housing where they are positioned by a spacer. Pins extend through the spacer to grooves in the bearing shell to prevent rotation. Radial loads on the bearings are 600 to 800 pounds. Thrust loads are small due to the "floating" action of the pinions.

The main and auxiliary racks move vertically through long, unsheathed grade 14 Graphitar bearings. The main rack bearing is 4.5 in. long and the auxiliary rack bearing is 6.5 in. long. One side of the central portion of each bearing is machined away to permit pinion-to-rack engagement. The long bearing surface offers excellent support to the rack from pinion thrust loads. Unlike the pinion shaft bearings, the rack Graphitar bearings are shrunk into the housing.

9.7.4.2 Support Tube Assembly. The support tube assembly consists of the rack and pinion housing, described above, the lower main and auxiliary tubes which screw into the housing, and an upper main tube that is threaded into the top of the housing. The weight of the total assembly is approximately 175 pounds. The overall length of the assembly is about 145-1/2 in.

The upper main support tube is a 40.8 in. long, 2-9/16 in. O.D. heavy walled tube of type 304 stainless steel. The inside diameter is 1.625 in., except for a 1-1/2 in. diameter threaded hole at the upper end which receives the shock support bolt. The tube wall is pierced with a number of 1/2 in. holes for free circulation of the heavy water. The tube is threaded into the rack and pinion housing and is prevented from unscrewing during operation by angle clips. The clip fasteners are tack welded in place.

A latch assembly is mounted on the upper main support tube. One element of the latch assembly is the hard stop against which the main rack collar is driven at the upper limit of the 30 in. normal stroke. If the hard stop is rotated out of the way the rod can be driven upward an additional 6 inches, where the rack collar engages the upper latch. The changing of the blades is done with the rack assembly in this raised position. The rack collar cannot be disengaged from the upper latch until the rod is raised slightly to free the latch and the latch is manually rotated out of the way. Both the upper latch and the hard stop are spring-loaded to return to normal operating position.

Below the latch assembly a bracket is welded to the upper support tube for use with a lifting tool during installation or removal of the drive. A projection of the bracket extends over the auxiliary rack centerline and serves as an

upper hard stop for the auxiliary rod. The bracket and latch assembly are both of 304 stainless steel.

The lower main support tube is 81-1/8 in. long, 2-1/8 to 2-1/4 in. O.D., and 1.750 in. I.D. The material is type 304 stainless steel. The tube is threaded into the rack and pinion housing at the upper end and has the snubber assembly threaded to it at the lower end. The tube is prevented from unscrewing by the neutron shield which is screwed and tack welded to it and to the dowel pin plate. A series of 5 collars spaced along the length of the tube serve as stand-off supports for the neutron shield. The central section of the tube is pierced with a series of 11/16 in. diameter holes which allow heavy water to escape freely as the rack piston moves downward in the first part of the scram stroke. The lower portion of the tube is not pierced, and acts as part of the dashpot in a scram.

The lower auxiliary support tube is 108-1/8 in. long, 1.366 to 1.75 in. O.D., and 1.125 in. I.D. The material is type 304 stainless steel. The tube is threaded into the rack and pinion housing at the upper end. Angle clips with tack welded fasteners prevent the tube from unscrewing during operation. There are longitudinal slots in the central portion of the tube through which the auxiliary rack is connected by pins to an external collar. The collar restrains the upper end of the auxiliary rod spring, and slides up and down along the tube with the rod motion. The lower end of the spring is restrained by a retainer plate which is bolted to the dowel pin plate. At the bottom of the tube an integral cylindrical bearing guides the auxiliary rack. The bearing is chrome plated for wear resistance, as is the outside of the support tube where the spring collar slides.

9.7.4.3 Shock Support Bolt and Lateral Guides. The support tube assembly (previous section), which is the stationary portion of the internal drive mechanism, is hung from the reactor vessel cover by the shock bolt assembly. The support tube assembly is attached to the flow shroud just below the pinion housing by the lateral guide plate, and again at the lower end of the tube assembly by the dowel pin plate. Both the latter attachments permit limited vertical motion of the tube assembly to accommodate thermal expansion, but provide rigid horizontal support. All of these mounting structures are of type 304 stainless steel, with chrome plating on wear surfaces.

The shock bolt assembly is the fixed point of attachment of the internal drive mechanism to the reactor vessel. The assembly consists of the bolt, a mating pair of eccentric bushings, a shim, and a lock washer assembly. The shim is a spacing collar which fits between the top of the upper main support tube and the under side of the stainless steel reactor vessel cover. Shims are individually fitted at the time of mounting of the drive mechanisms to obtain uniform vertical positions. The eccentric bushings fit in a recess in the upper side of the vessel rod mounting flange. The purpose of the bushings is to provide a small degree of lateral freedom in mounting a drive, to allow a precision alignment with the flow shroud members. The bolt passes through the eccentric bushings, a hole in the vessel mounting flange, and the shim, and threads into the upper main support tube. The bolt itself is a hex-head type, 1-1/2 in. diameter by 9-5/8 in. long. When the bolt is turned down tight it draws the upper main support tube and shim hard against the vessel cover. The lock washer assembly prevents the bolt from unscrewing and locks the eccentric bushings in the correct position.

The lateral guide plate is rigidly attached to a mounting plate on the top flange of the flow shroud (elevation 130 ft-7-3/4 in.) by spring-loaded pins. The guide plate includes a precision-bore collar which fits around the lower

main support tube just below the rack and pinion housing. The support tube slides vertically in the guide plate collar for thermal expansion take-up. The diametral clearance of the collar and the main support tube is 0.005 in. The differential expansion between the stainless steel drive structure and the aluminum vessel for the shutdown to normal operation temperature range (70-120°F) is 0.013 in. at the lateral guide plate. The effect of the differential expansion is an upward motion of the drive with respect to the vessel as the temperature rises.

The dowel pin plate is attached to the lower main support tube and the lower auxiliary support tube at elevation 124 ft - 5-1/2 in., just above the lower end of the main tube. The main tube is a slip fit in a hole in the dowel pin plate and is locked to the plate by the neutron shield. The auxiliary support tube is attached to the plate by a snug-fitting retainer plate which is bolted to the dowel pin plate. The dowel pin plate is attached to the flow shroud by two ball pins which slide in close-fitting holes in blocks welded to the shroud. Lateral motion of the support tubes with respect to the flow shroud and with respect to each other is thus prevented, while the necessary vertical expansion motions can take place without restraint. The shutdown to operating condition temperature range results in a total differential expansion at the dowel pin plate level of 0.033 in. The effect of the expansion is an upward motion of the drive with respect to the vessel as the temperature increases.

9.7.4.4 Neutron Shields. The function of the neutron shields in attenuating the thermal neutron flux in the neck of the reactor vessel is discussed in Section 10.3.4. A neutron shield is mounted on the lower main support tube of each drive mechanism. The shield is a half-tube 77 in. long, 2.750 in. I.D., and 0.200 in. total wall thickness. The shield is attached to the main support tube on the side toward the vessel center at four collars on the tube, one collar on the snubber cylinder, and at the dowel pin plate. Three stainless steel machine screws hold the shield to each collar. A stainless steel clamping strip serves as a common washer for the three screws at each collar. The screw heads are tack welded to the clamping strips to prevent loosening. The shield extends from elevation 124 ft-0 in. to elevation 130 ft-5 in., covering nearly the whole length of the lower main support tube and snubber cylinder.

The shield contains a 0.040 in. thick layer of cadmium completely framed and clad with 304 stainless steel. The final welding of the shield assembly is done in an evacuated chamber to insure that no gas is trapped in the plate. The cadmium is bonded to the stainless steel cladding by prolonged heating after the welding is finished. A final autoclave test is used to check for trapped gas. The shields were manufactured by Dresser Products, Inc., of Great Barrington, Mass.

9.7.4.5 Main Rack Assembly. The main drive rack assembly consists of a rack with a collar at the upper end, a coil spring, a piston, a snubber assembly, a rack follower, and a main blade female coupling. With the exception of the snubber, the rack assembly includes all of the internal drive line components which move vertically to position the main blade. The snubber is attached to the main lower support tube in assembling the mechanism, and is stationary. It is included here with the rack assembly because it is captive on the rack follower and is removed with the rack assembly. The main rack assembly weighs 42 lb and is about 142 in. long. Type 304 stainless steel is used for all components except a Graphitar bearing and the coil spring, which is Inconel X. Stainless steel wear surfaces are chrome plated.

The rack is a 1 in. nominal diameter solid bar 86-1/2 in. long. A 40.39 in. section of the bar near the upper end has 90 teeth of 7 diametral pitch, 20°

stub involute form cut into it. The rack teeth have a face width of 0.781 in., and match the main pinion teeth. The corners of the rack teeth are rounded so that they do not score the coil spring. Flat tracks are cut in the bar on both sides of the rack teeth to engage the main pinion side plates. The rack bar is threaded at both ends for attaching the rack collar at the upper end and the piston at the lower end. In assembling the internal drive mechanism the rack assembly is slid into the support tube assembly (less the upper support tube) until the upper end of the rack protrudes above the pinion housing. The main rack collar is then screwed onto the rack and pinned in place. The collar is 1.615 in. O.D. and 1.250 in. long.

The piston is 6-1/2 in. long and 1.7320 in. O.D. It is internally threaded at both ends to receive the lower end of the rack and the upper end of the rack follower. A series of 6 circumferential grooves, 1/2 in. long and 0.085 in. deep, are cut on the piston to form a pressure-breakdown leakage annulus in the 1.750 in. I.D. of the lower main support tube. The threaded connections of the rack and rack follower to the piston are locked by pins after assembly.

The coil spring rides on the rack between the piston and the pinion housing, inside the lower main support tube. As the drive train moves upward in withdrawing the main blade from the core the coil spring is compressed (see Section 9.7.2.4). Data on the spring are given in Table 9.7-5.

The rack follower is a 1.001 in. diameter bar 53-3/16 in. long. The upper end is threaded for connection to the piston. The lower end is enlarged in diameter to 1.420 in. for a length of 2.187 in. and has a slot and a 20° female taper cut in it as part of the blade coupling. A coupling nut is held captive on the follower shaft, and a flat-sided collar above the nut is fixed to the shaft to provide a holding point for the blade coupling tool. Details of the coupling are shown in Figure 4.8.

Table 9.7-5 Main Drive Internal Coil Spring Data

Material	Inconel X
Wire diameter	0.172 in.
Coil inside diameter	1.208 in.
Number active coils	194
Free length	75 in.
Preload	4 in.
Spring rate	3.6 lb/in.
Deflection, full-out rod	34 in. (including 4 in. preload)
Stress, full-out rod	80,000 psi
Force, full-out rod	130 lb
Force-displacement curve	Figure 9.29

The snubber cylinder has a pressure-breakdown section and a lower bearing

for the rack follower. The snubber is 8-1/2 in. long and 2-1/4 in. in diameter. The rack follower passes through the snubber. The pressure-breakdown section consists of a series of 6 circumferential grooves and lands in the hole through the snubber. The grooves are 1/2 in. long and 0.126 in. deep. The groove at the lower end of the group is connected to the water volume outside the drive mechanism by four 1/2 in. diameter holes. The lower bearing for the rack follower is of grade 14 Graphitar. The Graphitar tube is a shrink fit in the snubber. The snubber is attached to the end of the lower main support tube by a threaded connection. A collar on the snubber supports the lower end of the neutron shield, and the attachment of the shield to the snubber and support tube prevents the snubber from unscrewing.

9.7.4.6 Auxiliary Rack Assembly. The auxiliary rack assembly consists of a rack, an upper collar, and an auxiliary blade coupling. The assembly is 147-1/8 in. long and weighs 34 lb. All components are of type 304 stainless steel and wear surfaces are chrome plated.

The rack is a 1 in. diameter solid bar 143 in. long. It is threaded at the upper end for the collar and at the lower end for the blade coupling taper piece. A 19.474 in. long section of the bar near the upper end has 44 teeth of 7 diametral pitch, 20° stub involute form cut into it. The teeth have a face width of 0.781 in. Flat tracks are cut in the bar on both sides of the teeth to engage the auxiliary pinion side plates.

The upper collar is screwed onto the upper end of the rack and is pinned in place after the correct position is determined. The collar provides a means of adjusting the upper stop position for the auxiliary drive train.

The auxiliary blade coupling is similar to that for the main blade. The auxiliary rack shaft ends in a female taper piece which is screwed into the rack and pinned in place. A coupling nut is captive on the rack and a fixed, flat-sided collar provides a holding point for the blade tool (see Figure 4.8).

The rack shaft has a transverse 0.191 in. diameter hole located 72-17/32 in. from the lower end for the pin which connects the rack to the sliding auxiliary spring collar.

9.7.4.7 Auxiliary Spring. The auxiliary spring is a coil spring located on the lower auxiliary support tube. It is captive between an upper collar which is pinned to the auxiliary rack through slots in the support tube, and the fixed retainer plate bolted to the dowel pin plate. The spring keeps an upward force on the auxiliary drive train to remove backlash in the gearing and to insure insertion of the auxiliary rod if a break should occur in the drive train. Spring data are given in Table 9.7-6.

Table 9.7-6 Auxiliary Spring Data

Material	Inconel X
Wire diameter	0.187 in.
Coil inside diameter	1.406
Number active coils	105
Free length	55 in.

Table 9.7-6 (cont'd)

Solid height	20 in.
Spring rate	4 lb/in.
Preload	17-1/2 in.
Deflection, full-out rod	32-1/2 in. (incl. 17-1/2 in. preload)
Stress, full-out rod	96,500 psi
Force, full-out rod	130 lbs
Force, preload	70 lbs

9.7.5 POSITION INDICATOR. The control rod position indicator provides the Operator in the control room with a continuous display of main and auxiliary rod positions. The rod positions are graphically shown by 16 film strips, one for each blade. The eight film strips representing the main rods are placed side by side at the top of the display panel and move upward as the main rods are withdrawn. The eight film strips for the auxiliary rods are located below the main rod strips and move downward as the auxiliary rods are withdrawn. The scale of the indicator display is the same as that of the rod motions: one inch of rod motion gives a one inch motion of the corresponding film strip. The indicator display thus reproduces the rod motions and positions just as they occur in the core region. The relative positions of the rods in the main or auxiliary crowns are immediately apparent to the Operator without any need to interpret dial readings or numerical data.

The position indicator panel is 19 in. wide, 21 in. deep, and 70 in. high. Windows in the front panel display the eight 30 in.-long main film strips and the eight 15 in.-long auxiliary film strips. Rod position scales with a least division of 0.1 in. are engraved on the display panel between each film strip. Zero rod position for both main and auxiliary rods is at the "full-out" position.

Above the main and below the auxiliary film strips are indicator lights, a set of three for each blade. The outermost light, marked "OUT", indicates full blade withdrawal and is red in color. The inner light, marked "IN", indicates full blade insertion, and is green in color. The intermediate light in each set is amber and is lit when the rod is neither full-in or full-out. The lights are connected to position limit switches on the external drive packages.

Each of the 16 film strips is an endless belt as shown in Figure 9.28. The strips are made from 0.004 in. thick DuPont "Cronar" 35 mm movie-type film. Half of the film is colored red to simulate the blade while the remaining half is colored white for contrast. Film sprockets are mounted on the auxiliary torque receiver shafts and on shafts geared to the main servo motor drives. Idler sprockets assure a proper wrap of the film around the driving sprockets. The film does not stretch with age and is not affected by normal temperature and humidity changes. After initial tightening of the strips further belt takeup is not expected to be necessary.

The auxiliary rod position indicator drive is of the standard synchro type. A Bendix torque transmitter is geared directly to the mechanism drive shaft and

turns 4.99 revolutions for the 15 in. blade stroke. A magnetic field pattern is transmitted by the transmitter to a receiver in the position indicator. The receiver rotor is also energized, producing another magnetic field, and a torque results which tends to align the two fields. Since the receiver is electrically identical to the generator, a direct torque transmission results. The torque gradient is small enough and the speed slow enough in the auxiliary drive to prevent excessive errors or slippage. A film sprocket is mounted on the receiver shaft to actuate the auxiliary film strip.

The main drive position indicator must follow the scram motion since the transmitter is geared to the drive shaft on the reactor side of the clutch. A servo system is used to drive the main rod indicator, to avoid the slippage and loss of phase which would be suffered by a multi-turn torque transmission system. The servo system consists of a Daystrom servo transmitter geared to the main drive shaft, a control transformer which receives the transmitter signal, a servo amplifier, and a servo motor. Figure 9.28 gives the manufacturer's model number for these units. The transmitter is energized from a 6.3 VAC source. The transmitter produces a voltage proportional to its shaft (i.e., rotor) angular position. As noted in Section 9.7.3.5 the transmitter shaft turns 0.437 revolutions for the 30 in. main rod stroke, and is thus free of ambiguity in the number of turns in a fast motion of the rod. The control transformer is also a rotating shaft unit and is geared to the film strip drive. The difference in transmitter and receiver rotor positions generates an error signal which is amplified to a level sufficient to operate a 2 phase AC servo motor. The servo motor is geared to the film strip drive sprocket shaft. The servo motor turns in the direction necessary to reduce the error voltage and thereby drives the film strip. Although the main rod indicator will not slip phase in the scram motion, it is not fast enough to follow the rapid motion as it occurs. The main rod moves 30 in. in less than 0.7 seconds in a scram from the full-out position, with the first half of the stroke occurring in less than 0.2 seconds. The main position indicator lags the scram motion by several seconds, but gives a correct rod position as soon as it catches up with the rod.

The accuracy of the position indication for both main and auxiliary rods is ± 0.100 in. of true rod position. The reference points in the motions of the rod drive trains are the upper hard stops.

The position indicator panel receives electrical power at 115 VAC from the instrument electrical panel in the control room. The position indicator thus receives power from the emergency generator set through this panel should the normal commercial source fail. In the event of a further electrical fault which removes the emergency generator power, the rod position lights for "IN", "OUT", and intermediate positions are unaffected since they are continuously connected to a battery power supply (the 24 VDC instrument battery, see Section 9.8). The position indicator panel contains the power supplies for the 6.3 VAC supply to the main rod indicator servo generator and motor, and the 28 VDC regulated supply to the main rod indicator servo amplifier.

9.7.6 TESTING. An extensive test program was carried out on the prototype drive mechanism to demonstrate the adequacy of the design. Tests were made on individual components and on the assembled mechanism. Life tests were carried out at 140°F and 275 psig, in demineralized light water. After some minor design changes the mechanism was found satisfactory in all respects. The production mechanisms for the reactor were tested in the same manner, but underwent fewer cycles of full travel and scram. In tests on the assembled mechanism, the mounting arrangements in the reactor vessel were reproduced exactly in the test vessel and dummy control rod blades were coupled to the mechanism.

Test results are summarized in Tables 9.7-7 and 9.7-8. Reference 9.9 is the source of this material.

Table 9.7-7 Prototype Mechanism Component Tests

Power spring	Force-displacement diagram, Figure 9.29, and Table 9.7-4, Sec. 9.7.3.3			
Main coil spring	Force-displacement diagram, Figure 9.29, and Table 9.7-5, Sec. 9.7.4.5			
Auxiliary coil spring	Table 9.7-6, Sec. 9.7.4.7			
Belleville washers	0.065 in. deflection at 900 lb 0.750 in. deflection at 10,000 lb			
Motor and gear reducer torque tests				
	<u>Main Motor Output (in.-oz)</u>	<u>Aux. Motor Output (in.-oz)</u>	<u>Main Reducer Output (in.-lb)</u>	<u>Aux. Reducer Output (in.-lb)</u>
Rated torque	400	250	970	660
Measured torque at full speed	417	275	1050	1290
Measured stalling torque	524	320	1875	1825 (dn) 2150 (up)
Motor voltage variation	±10%, no effect, main or aux. +20%, no effect, main or aux. -20%, main motor pulled out of synchronism			
Clutch frictional torque	30 in.-lb			
Shaft seal frictional torque	60 in.-lb			
Clutch holding current	0.325 amp at 1200 in.-lb			
Clutch release time	0.012 sec, with filtering			

Table 9.7-8 Prototype Mechanism Tests, Assembled Unit at Service Conditions

Main follower lateral movement in 30 in. stroke	less than 0.015 in., any direction
Aux. follower lateral movement in 15 in. stroke	less than 0.015 in., any direction
Main rod speed variation, 30 in. stroke	4.31 - 4.36 in./min
Aux. rod speed variation, 15 in. stroke	4.22 - 4.29 in./min
Stall at upper and lower hard stops, both rods	no effect, extended time stalling

Table 9.7-8 (cont'd)

Stop, start, reverse time tests:	Main Rod <u>(sec)</u>	Aux. Rod <u>(sec)</u>
Stopping time, max.	0.0404	0.030
Starting time, max.	0.160	0.250
Reverse time, max.	0.240	0.320
Position indicator accuracy tests:	Main Rod <u>(in.)</u>	Aux. Rod <u>(in.)</u>
Deviation, film strip	+0.031	+0.078
position from rod	to	to
position	-0.078	-0.078
Shaft seal leakage in life test	4 cm ³ /hr, main 0, aux.	
Jogging test: 8 hr. duration, motors on 1.5 sec every 3 sec	no effect, main or aux. drives	
High speed moving pictures of scram motion	no lateral whip or vibration of blade	
Scram time tests from several initial positions	Figure 9.31	
Strain gage tests during scrams:		
<u>Point</u>	<u>Max. Stress (psi)</u>	<u>Force (lb)</u>
Lower main support tube between relief holes	1155	534
Upper support tube	770	2220
Belleville washer stack		886
Thread relief section, rod follower	860	534
Shock bolt, thread undercut	1565	2220
Rack, thread relief section	1710	886
Rack, tooth section	1912	886
Life and wear test:		
1300 driven cycles, main and auxiliary rods	no appreciable wear or malfunction, no appreciable change in scram characteristics	
120 scrams at 30 in.		
100 scrams at 25 in.		
100 scrams at 20 in.		
50 scrams at 10 in.		
50 scrams at 5 in.		
Automatic control test, auxiliary rod number 1:		
Maximum deviation of differ- ential transformer mechanical linkage	+0.365% to -0.186% of 0.065 in. range	
Dead band	9 μ amp of 1-5 mamp range	

Table 9.7-8 (cont'd)

Maximum deviation of blade position from nominal	+0.48% to -0.91% of 15 in. range
--	-------------------------------------

9.8 INSTRUMENT AND CONTROL ROD POWER9.8.1 NORMAL INSTRUMENT AND CONTROL ROD POWER

9.8.1.1 AC Power. The main and auxiliary control rod drive units, including the main rod clutches, are powered from the Control Rod Power Panel in the control room. This panel, in turn, is powered from the normal building AC distribution system through MCC #2 on the operations level (see Section 3.5.6.10). No emergency power source is provided for the control rod drive units.

All AC powered instrument components are supplied from the Instrument AC Power Panel in the control room. This panel, in turn, is powered from the Emergency Motor Control Center (see Section 3.6.5.9). The Emergency Motor Control Center receives power from the building 440 VAC system in normal conditions, and from the emergency generator set when the normal power fails.

9.8.1.2 DC Power. The DC powered instrument components are supplied from the 250 VDC battery banks and from the 24 VDC instrument emergency power supply batteries.

The 250 VDC batteries consist of two banks of lead-acid cells, each bank rated at 250 V with a 3 hour discharge capacity of 249 amp-hours. The batteries are located in the Graphite Reactor Fan House and are kept fully charged by motor generator sets at that location. The batteries are shared with the Graphite Reactor. Table 9.8-1 summarizes the total load connected to these batteries.

Table 9.8-1 250 VDC Fan House Battery Loads

Continuously connected loads:

HCe-101C, siphon break valve solenoid	15	watts
HCe-102, depressurization valve solenoid	12	
CV-IB-303F, inlet air valve solenoid	15	
CV-IB-304F, inlet air valve solenoid	15	
CV-IB-306F, exhaust air valve solenoid	15	
IB-304 condensate drain valve solenoid	15	
IB-306 condensate drain valve solenoid	15	
K-115, exhaust blower stop relay	15	
K-114A, nuclear incident alarm relay	15	
K-115B, nuclear incident alarm relay	<u>15</u>	
TOTAL, continuously connected load	147	

Loads connected during an AC power failure:

Continuous loads, listed above	147	watts
HFBR intercommunication system	650	
HFBR emergency lights	15000	
HFBR shutdown pump GA-102A or B	11200	
Total Graphite Reactor load	<u>23500</u>	
TOTAL, connected load in power failure	50497	

The 24 VDC instrument emergency power supply battery is a nickel cadmium alkaline storage unit rated at 24 volts with an 8 hour discharge capacity of 46 amp-hours. The battery is located in the instrument shop area immediately below the control room and is kept fully charged by a battery charging circuit. Table 9.8-2 summarizes the total load connected to these batteries.

Table 9.8-2 24 VDC Instrument Emergency Battery Loads

Continuously connected loads:

K-107A, B, and C, low-low level relays	7.2 watts
K-108A and B, pump stop relays	24
K-112A, B, and C, low-low pressure relays	7.2
K-113A and B, pump stop relays	24
Control rod position indicator limit lights	<u>32</u>
TOTAL, continuously connected load	94.4

Loads connected during an AC power failure:

Continuous loads, listed above	94.4 watts
FRa-101, shutdown flow indicator	15
Nuclear channel #5 indicator	40
Nuclear channel #6 indicator	<u>80</u>
TOTAL, connected load in power failure	229.4

A 48 VDC battery supply is provided for the tripping switchgear in the HFBR 2300 and 440 VAC systems. The battery is a nickel cadmium alkaline storage unit rated at 48 volts with an eight hour discharge capacity of 20 amp-hours. The battery is kept fully charged by a charging circuit. The battery and charging unit are located in cubicle #5 of the 2300 volt switchgear. The continuously connected load on the 48 volt battery is 10 watts, due to the switchgear monitoring lights. The intermittent switching loads are due to the trip coils in 10 breakers. The battery capacity is sufficient to operate the trip coils for an extended period following an AC power failure.

9.8.1.3 Instrument Air and Helium Supply. Instrument compressed air is supplied from two compressors and a receiving tank located on the equipment level (see Section 3.5.5). The receiver pressure is about 90 psig, and distribution throughout the building is made at that pressure. Pressure reducing valves at each pneumatic instrument reduce the air pressure to 15 psig or less for instrument use. One of the compressors is powered from the emergency generator set in the event of a failure of the normal power.

Helium for instrument use is supplied from either a low pressure manifold at 50 psig or a high pressure manifold at 300 psig. Local pressure reducing valves reduce the gas pressure as required for instrument use. The manifolds are supplied from compressed helium cylinders. A minimum supply of several days of normal reactor consumption is kept in the cylinder banks in the HFBR building. Sufficient helium for extended periods is immediately available from the laboratory compressed gas warehouse stocks.

9.8.2 INSTRUMENT EMERGENCY POWER. In the event of a failure of the normal AC power, the emergency generator starts automatically. All instruments are thus fully powered as long as the emergency generator operates. A number of instruments are powered by batteries during the time the generator set takes to start and reach full speed. These batteries also keep the instruments operating for a limited time if the generator set fails to start. The battery systems and the instruments they supply are listed in the sections which follow.

9.8.2.1 18 VDC Local Instrument Batteries. The instruments equipped with internal batteries are listed in Table 9.8-3. The internal batteries are nickel-cadmium cells rated at 17.6 VDC for 25 ma-hours, except for batteries in LRa-101 and LIa-106, which are rated at 17.6 VDC, 0.5 amp-hours.

Table 9.8-3 Instruments with Internal 18 Volt Batteries

<u>Instrument</u>	<u>Reserve Period</u>
TRa-101, core inlet temperature	8 min
TR-102, core outlet temperature	8
PRCa-101, reactor cover gas pressure	10
PdRa-101, core differential pressure	9.5
FRa-100, primary flow rate	3.3
FRa-101, shutdown flow rate	12
LRa-101, reactor liquid level	120
LIa-106, reactor liquid level	120
FRa-201, thermal shield coolant flow	11
Nuclear incident alarm system units	5

9.8.2.2 24 VDC Instrument Emergency Power Supply. Instruments connected to the 24 VDC battery system, described in Section 9.8.1.2, are listed in Table 9.8-2. Under full emergency load the reserve period in the 24 volt battery bank is about 3.6 hours.

9.8.2.3 12 VDC Radiation Instrument Emergency Power Supply. These batteries supply power to the radiation monitoring channels and the associated equipment. A single battery charger maintains the charge on 5 batteries. Upon loss of AC power, emergency battery power is supplied through 5 circuits, each having a battery, a rotary inverter, and a "flip-over" relay. The batteries are nickel-cadmium alkaline cells rated at 12 volts with an 8 hour discharge capacity of 46 amp-hours. The inverters are rotary units rated at .21 KVA at 115 VAC 60 cycle output. Table 9.8-4 summarizes the total load connected to these batteries.

Table 9.8-4 12 VDC Radiation Monitoring Instrument Emergency Battery Load

<u>Circuit</u>	<u>Instrument</u>	<u>Load</u>	<u>Reserve Period</u>
1	10 Channel Area Monitor	210 watts	2 hours
2	4 Channel Secondary Water Monitor	70	5
3	2 Channel Fission Product Monitor	60	5
4	2 Channel Stack Gas Monitor (measuring circuits only)*	60	5
5	2 Channel Nuclear Incident Monitor	85	3.6

* The stack gas monitor pumps are powered from the emergency motor control center.

9.8.2.4 250 VDC System. The load connected to this system is shown in Table 9.8-1. The batteries are described in Section 9.8.1.2. Under emergency conditions the reserve period of the batteries at full emergency load is approximately 2-1/3 hours.

References

- (9.1) R. E. Meagher and E. P. Bently, Rev. Sci. Instr., 10, 336 (1939).
- (9.2) W. H. Jordan, H. B. Frey, and G. Kelley, An Instrument for Measuring the Logarithms of Neutron Level and the Period of a Pile, ORNL-110 (Nov., 1948).
- (9.3) D. G. Pitcher, "Period Circuit Trip Times", BNL Memorandum (April 7, 1964).
- (9.4) W. Lones, Ionization Chambers for Reactor Operation, BNL 561 (May, 1959).
- (9.5) P. R. Tichler, "Temperatures and Burnout Margins at Various Instrument Trip Levels", BNL Memorandum (January 2, 1964).
- (9.6) J. M. Hendrie, "HFBR Control Rods", BNL Memorandum (January 9, 1961).
- (9.7) C. Andrea, Preliminary Design Report, Diamond Power Specialty Corp. (March 19, 1962).
- (9.8) Design of a Snubber, Babcock and Wilcox Co. Research Center, Alliance, Ohio (March 8, 1962).
- (9.9) C. Andrea and J. R. Thayer, Prototype Test Report, Diamond Power Specialty Corp. (March 22, 1963), and Supplement No. 1, Prototype Test Report, Diamond Power Specialty Corp. (June 28, 1963).

Figure 9.1 Arrangement of the reactor control room.

Figure 9.2 Layout of the control console.

SPARE	D ₂ O LEVEL <u>HI-LOW</u> <u>REACTOR</u> LRa-101	DIFF PRESS <u>HI-LOW</u> <u>REACTOR</u> PRa-101	INLET TEMP <u>HI-LOW</u> <u>PRI. COOL.</u> TRa-101	H ₂ O COOLANT <u>TEMP LOW</u> <u>HX A</u> TRa-7	D ₂ O LEAK <u>SHUT DOWN</u> <u>SYSTEM</u> RRa-302	ABNORMAL <u>CONDITION</u> <u>EMER. GEN.</u> GEN. ANN.	RADIATION <u>HI-LOW</u> <u>BALCONY</u> NUC. INCID.
SPARE	PRI-SEC <u>D/P LOW</u> <u>HX A</u>	PRESS <u>HI-LOW</u> <u>REACTOR</u> PRCa-101	LOW FLOW PRIMARY <u>COOLANT</u> FRa-100	H ₂ O COOLANT <u>TEMP LOW</u> <u>HX B</u> TRa-8	FLOW LOW <u>SHUT DOWN</u> <u>SYSTEM</u> FRa-101	FLOW LOW SECONDARY <u>COOLANT</u> FRa-300	LEVEL HI-LO TH. SHIELD <u>EXPAN TANK</u> LIa-201
SPARE	PRI-SEC <u>D/P LOW</u> <u>HX B</u>	D ₂ O LEAK PRIMARY <u>SYSTEM</u> RRa-301	<u>CONDUCT HIGH</u> PRIMARY <u>COOLANT</u> Cra-2	H ₂ O COOLANT <u>TEMP LOW</u> <u>SUPPLY</u> TRa-9	SEC. FLOW LOW <u>SHUT COOL</u> FIa-301	COOLANT TEMP HIGH <u>TH. SHIELD</u> TRa-6	LOW FLOW THERMAL <u>SHIELD</u> FRa-201

COOLANT <u>FLOW LOW</u> <u>THIMBLE 10</u> FIa-10	COOLANT <u>FLOW LOW</u> <u>THIMBLE 13</u> FIa-13	COOLANT <u>FLOW LOW</u> <u>THIMBLE 16</u> FIa-16	TEMP HI-LO <u>EXPER</u> <u>FACILITIES</u> TRa-4	LEVEL HI-LO <u>EXPER</u> <u>COOLER</u> LIa-103	SH. DOOR <u>OPEN</u> <u>SH.DN.COOL.</u>	<u>PRESS LOW</u> <u>HE SUPPLY</u> PIa-201	<u>PRESS LOW</u> INST AIR PA-301
COOLANT <u>FLOW LOW</u> <u>THIMBLE 11</u> FIa-11	COOLANT <u>FLOW LOW</u> <u>THIMBLE 14</u> FIa-14	LOW FLOW SHUTTER <u>COOLANT</u> FIa-203	<u>FLOW LOW</u> PLUG <u>COOLANT</u> FIa-204	D ₂ O LEAK <u>TO ROOM</u> <u>EQ LEVEL</u> CAB 1	SH. DOOR <u>OPEN</u> <u>HX A</u>	<u>PRESS LOW</u> HE SUPPLY PIa-202	LEVEL HI-LO CANAL LA-307
COOLANT <u>FLOW LOW</u> <u>THIMBLE 12</u> FIa-12	COOLANT <u>FLOW LOW</u> <u>THIMBLE 15</u> FIa-15	<u>TEMP HIGH</u> SHUTTER <u>COOLANT</u> TRa-11	<u>FLOW LOW</u> CONCRETE <u>COOLANT</u> FIa-205	D ₂ O LEAK <u>TO ROOM</u> <u>OP LEVEL</u> CAB 2	SH. DOOR <u>OPEN</u> <u>HX B</u>	LEVEL HI-LO BIO SHIELD <u>EXPAN. TANK</u> LIa-203	48 VDC FAILURE SWITCHGEAR

Figure 9.3 Front face display of the annunciator panel for "alarm-only" signals (panel located above instrument panels 1, 2, and 3).

POWER RISE <u>UNSAFE</u> <u>REACTOR</u> SAF AMP 1	DIFF PRESS <u>HIGH-LOW</u> <u>REACTOR</u> PdRa-101	PRESS LOW <u>REACTOR</u> PRCa-101	<u>LOW FLOW</u> PRIMARY <u>COOLANT</u> FRa-100	D ₂ O LEVEL <u>LOW</u> <u>REACTOR</u> Lra-101	REACTOR <u>SHUT DOWN</u> <u>CONSOLE</u> MAN TRIP 1	REACTOR <u>SHUT DOWN</u> <u>N EX LEVEL</u> MAN TRIP 4	REACTOR <u>SHUT DOWN</u> <u>S EQ LEVEL</u> MAN TRIP 7	SPARE
POWER RISE <u>UNSAFE</u> <u>REACTOR</u> SAF AMP 2	DIFF PRESS <u>HIGH-LOW</u> <u>REACTOR</u> PdA 102	<u>PRESS LOW</u> <u>REACTOR</u> PA-102	<u>PUMP A OFF</u> PRIMARY <u>COOLANT</u> BKR	D ₂ O LEVEL <u>LOW</u> <u>REACTOR</u> LA-102	REACTOR <u>SHUT DOWN</u> <u>N OP LEVEL</u> MAN TRIP 2	REACTOR <u>SHUT DOWN</u> <u>S EX LEVEL</u> MAN TRIP 5	SPARE	SPARE
POWER RISE <u>UNSAFE</u> <u>REACTOR</u> SAF AMP 3	DIFF PRESS <u>HIGH-LOW</u> <u>REACTOR</u> PdA 103	<u>PRESS LOW</u> <u>REACTOR</u> PA-103	<u>PUMP B OFF</u> PRIMARY <u>COOLANT</u> BKR	D ₂ O LEVEL <u>LOW</u> <u>REACTOR</u> LIa-106	REACTOR <u>SHUT DOWN</u> <u>S OP LEVEL</u> MAN TRIP 3	REACTOR <u>SHUT DOWN</u> <u>N EQ LEVEL</u> MAN TRIP 6	SPARE	SPARE

(SCRAM ANNUNCIATOR)

POWER RISE <u>UNSAFE</u> <u>REACTOR</u> SAF AMP 1	INLET TEMP <u>HIGH</u> <u>PRI. COOL.</u> TRa-101	SPARE	<u>LOW FLOW</u> THERMAL <u>SHIELD</u> FRa-201	<u>LOW FLOW</u> SHUTTER <u>COOLANT</u> FIa-203	D ₂ O LEVEL <u>LOW-LOW</u> <u>REACTOR</u> Lra-101	PRESSURE <u>LOW-LOW</u> <u>REACTOR</u> PRCa-101
POWER RISE <u>UNSAFE</u> <u>REACTOR</u> SAF AMP 2	INLET TEMP <u>HIGH</u> <u>PRI. COOL.</u> TIa-103	<u>PUMP A OFF</u> SHUT DOWN <u>SYSTEM</u> BKR	<u>PUMP A OFF</u> THERMAL <u>SHIELD</u> BKR	<u>PUMP A OFF</u> SHUTTER <u>COOLANT</u> BKR	D ₂ O LEVEL <u>LOW-LOW</u> <u>REACTOR</u> LA-102	PRESSURE <u>LOW-LOW</u> <u>REACTOR</u> PA-104
POWER RISE <u>UNSAFE</u> <u>REACTOR</u> SAF AMP 3	INLET TEMP <u>HIGH</u> <u>PRI. COOL.</u> TIa-104	<u>PUMP B OFF</u> SHUT DOWN <u>SYSTEM</u> BKR	<u>PUMP B OFF</u> THERMAL <u>SHIELD</u> BKR	<u>PUMP B OFF</u> SHUTTER <u>COOLANT</u> BKR	D ₂ O LEVEL <u>LOW-LOW</u> <u>REACTOR</u> LIa-106	PRESSURE <u>LOW-LOW</u> <u>REACTOR</u> PA-105

(SETBACK ANNUNCIATOR)

Figure 9.4 Front face displays of the annunciator panels for scram and setback signals (scram panel located above instrument panels 7 and 8, set-back panel located above instrument panels 9 and 10).

Figure 9.5 Block diagram of nuclear instrument channels, and reactor control and safety system.

Figure 9.6 Graphs of the relation between reactor positive period and power level for setback and scram, and for various values of the zero-power period for scram (τ_0).

Figure 9.7 Block diagram of a single nuclear safety channel.

Figure 9.9 Circuit diagram for the period amplifier.

Figure 9.10 Block diagram of the automatic power level control system.

Detector Thimbles And Chamber Positioners

Figure 9.11 Instrument tube and detector positioner details.

Figure 9.12 Process instrument channels for primary flow rate, primary coolant temperature, and primary power calculator.

Figure 9.13 Process instrument channels for primary system pressure.

Figure 9.14 Process instrument channels for core differential pressure and shutdown system flow rate and pumps.

Figure 9.15 Process instrument channels for primary coolant liquid level in the reactor vessel.

Figure 9.16 Process instrument channels for thermal and biological shield systems flow rates and pumps.

Figure 9.17 Scram system relay diagram. Relay contacts are shown in the "normal" position, i.e., with relay coils de-energized.

Figure 9.18 Setback system relay diagram. Relay contacts are shown in the "normal" position, i.e., with relay coils de-energized.

Figure 9.19 Rod control switch and relay diagram. Relay contacts shown in the "normal" position, i.e., with relay coils de-energized.

Figure 9.20 Flow reversal system relay diagram. Relay contacts are shown in the "normal" position, i.e., with relay coils de-energized.

Figure 9.21 Low-low pressure interlock relay diagram. Relay contacts are shown in the "normal" position, i.e., with relay coils de-energized.

Figure 9.22 Isometric view of the reactor vessel showing the mounting arrangement for a control rod drive mechanism.

Figure 9.23 Expanded isometric view of the main control rod drive train.

Figure 9.24 Details of the portion of the control rod mechanism inside the reactor vessel.

Figure 9.25 Isometric view of the lower ends of the control rod mechanism in the reactor vessel.

Figure 9.26 Drive shaft and shaft seal assembly drawing for the control rod mechanisms.

Figure 9.27 Plan view of the reactor shield pit at elevation 130 ft-8 in. showing the arrangement of the control rod drive mechanisms.

Figure 9.28 Schematic drawing of the rod position indicator system.

Figure 9.29 Contributions of the power spring and the internal coil spring to the total scram force of a main rod drive.

Figure 9.30 Scram snubber forces on the drive train as a function of time after deceleration begins.

Figure 9.31 Prototype drive mechanism scram test curves.

Figure 9.32 Diagram of the nuclear incident alarm system. Relay contacts are shown in the "normal" position, i.e., with relay coils de-energized.

SECTION 10. SHIELDING

10.1 SUMMARY DESCRIPTION

The basic shielding for the HFBR is provided by a lead and steel thermal shield and by a heavy concrete biological shield. The concrete structure forms a central column in the building from elevation 110 ft-11 in. to elevation 149 ft-0 in. The thermal shield and reactor vessel are located in a pit in the heavy concrete structure. The top of the pit is covered during normal operations by a heavy concrete plug. The location of these shield components in the reactor building is best seen in Figures 3.1 and 3.2. The Combustion Engineering, Inc. reports CEND-159, Parts III and IV (10.1, 10.2) are general references for the thermal and biological shields.

The pumps and heat exchangers for the primary system are located in cells immediately beneath the biological shield. There are three of these process equipment cells, with reinforced concrete walls to provide shielding. A heavy water storage tank and the heavy water purification system are located in pits below the equipment area floor, with ordinary concrete cover slabs for top shielding. Process equipment for various other auxiliary systems is shielded by stacked concrete blocks.

10.2 THERMAL SHIELD

10.2.1 LOCATION AND FUNCTION. The thermal shield is located in the base of the heavy concrete pit, as shown in Figure 10.1. The thermal shield rests upon a circular I-beam support member which is imbedded in heavy concrete. The support girder, in turn, rests upon the reinforced concrete structure of the experimental level. Immediately beneath the thermal and biological shields there are a series of walls for the equipment cells in the basement which serve as load transmitting members to the foundation.

A 1 in. gap around the sides and bottom of the thermal shield insures that thermal expansion is not restrained by the surrounding heavy concrete. The mounting pads on the thermal shield are radial steel plates which slide between guide bars on the support girder. This arrangement allows radial expansion of the shield but prevents rotation or linear motion. The thermal shield is maintained in a carbon dioxide atmosphere, both within the shield vessel and outside it in the radial and bottom gaps.

The primary function of the thermal shield is to protect the surrounding heavy concrete structure from excessive heating by the leakage radiation from the reactor. The heavy concrete used in the HFBR shield will accept an energy current density of 7.5×10^{11} Mev/cm²-sec (giving a maximum temperature rise of 50°F) without thermal stress cracking. The energy current from the active core is about 5×10^{13} Mev/cm²-sec at the shield surface, and the thermal shield must attenuate this current to the value above if the concrete is not to crack.

A second important function of the thermal shield is to support the reactor vessel. The reactor vessel is bolted to the upper thermal shield. The weight of the vessel is carried by the thermal shield and is transmitted to the foundations through the shield support ring.

Auxiliary functions of the thermal shield and its cover are to provide containment for heavy water leakage from the reactor vessel, or for any

accidents which might occur in the reactor vessel. The thermal shield cover (upper thermal shield) also acts as the primary shielding layer for the upper pit.

10.2.2 GEOMETRY, DIMENSIONS, AND MATERIALS. The thermal shield is composed of a multiple-walled cylindrical steel vessel and a three-component cover. A sketch of the assembled shield is shown in Figure 10.2. Details of the thermal shield vessel are shown in Figure 10.3, and details of the cover structure are shown in Figure 10.4.

The lower thermal shield is composed of a series of cylindrical steel shells, with water passages between some shells, and a shielding layer of lead between others. The composite wall is 9 in. thick, with 3 in. of steel plate, 1.25 in. of cooling water, and 4.75 in. of lead. The detailed structure of the side wall is shown in Figure 10.3. The bottom of the lower thermal shield is of the same type of construction, but with an increased water thickness due to the inner steel plate and flow baffle being dished heads of 7 ft-0 in. radius rather than flat plates.

The basic dimensions of the lower thermal shield are 9 ft-0 in. outside diameter, 7 ft-6 in. inside diameter, and over-all height 10 ft-6-3/4 in. The upper 26-3/4 in. of the shield vessel is simply an extension of the outer steel shell which serves as a tie ring for the hold-down bolts and wedges of the upper thermal shield. The centerline height of the interior cavity in the thermal shield is 7 ft-6 in. The lower thermal shield weighs 110,000 lbs, of which 70,230 lbs is due to the lead layer.

The cylindrical walls of the lower thermal shield are pierced by a series of holes which are opposite beam tubes in the reactor vessel. The various neutron beams emerge from the thermal shield cavity through these holes. Cooling water enters and leaves the lower shield through a pair of 8 in. pipes in the bottom. The inlet pipe is on the shield centerline, and the outlet pipe is 16 in. off-center. A 1.5 in. pipe (1-1/2 in. DV-120) runs inside the inlet pipe and penetrates the inner head of the shield. This line is the CO₂ inlet pipe to the shield cavity and is also the water drain from the cavity.

Four thermocouple wells extend the full height of the lower shield in the lead layer. These wells can be reached from the shielded cells in the equipment area during shutdown periods, so that thermocouples in them can be replaced as they go bad in the radiation field. A number of temporary thermocouples are attached to the thermal shield for use in start-up experiments, but are not expected to last more than a year.

The upper thermal shield is composed of three main pieces. The support ring is a toroidal box girder of rectangular cross section filled with steel shot and cooling water. The support ring provides substantial shielding, but its main purpose is structural support of the upper thermal shield assembly. The other components of the upper thermal shield are two split-ring sections. The assembled split-ring sections form a toroidal shield member with an L-shaped cross section. The base of the "L" is filled with layers of steel plates and cooling water passages. The vertical leg of the L is filled with steel shot and cooling water. The toroid is split on a vertical diametral plane to make the two split-ring sections. This complex construction is required so that the upper thermal shield may be assembled around the neck of the reactor vessel. Nozzles and flanges on the reactor vessel above the level of the upper thermal shield extend outside the diameter of the vessel neck, and prevent a simple, one-piece toroidal shield from being slid down over the completed vessel. The split-ring pieces, which are a fairly close fit to the

vessel neck, can be assembled under the nozzles and flanges. The support ring, which is a single complete ring unit, has a sufficiently large inside diameter so that it will pass over the reactor vessel flanges and nozzles.

The upper thermal shield sections are bolted together to form a single unit. The reactor vessel, in turn, is bolted to this assembly. The upper thermal shield rests on a flange at the top of the lower thermal shield, and is held in place there by a series of hold-down bolts and wedges which lock into the extended outer shell of the lower shield.

The basic dimensions of the upper thermal shield are 8 ft-9 in. outside diameter, 4 ft-4 in. inside diameter, and 1 ft-8 in. height. These dimensions are for the assembled shield rather than the component parts. The assembled upper thermal shield weighs 30,000 lbs. The upper thermal shield assembly is penetrated by three holes for the thermal irradiation thimbles. In addition, there is a hole for the installation of a thermopile assembly into the shield cavity.

The materials list for the upper and lower thermal shields is given in Table 10.2-1. The basic material of the upper and lower thermal shields is the SA-201 plate. This is a medium strength carbon-silicon steel used extensively for welded boilers and pressure vessels. In the Grade A Firebox Quality it has a tensile strength of 55,000 to 65,000 psi, a minimum yield point of 30,000 psi, and a minimum elongation in 2 in. of 29%. Bend test specimens must show no cracking on the outside of the bend after being bent cold through 180° around a mandrel of diameter equal to 1/2 the plate thickness for plates of 1 in. or less thickness, and of diameter equal to the plate thickness for plates 1 to 1-1/2 in. thickness. The brass pipe and fittings are used only in the thermocouple wells imbedded in the lead layer.

Table 10.2-1 Material List for the Thermal Shield

<u>Item</u>	<u>Material Specification</u>
Steel plate	SA-201, Grade A, Firebox Quality, with ultrasonic inspection as per SA-435
Steel pipe	SA-102, Grade A
Steel pipe fittings	SA-105, Grade I, or SA-106, Grade A
Steel tubing	SA-192, or SA-201, Grade A
Steel forgings	SA-181, with Si addition
Steel studs and pins	SA-193, B7
Steel nuts and washers	SA-194, Class 2H
Steel structural shapes	A-7
Steel balls	Soft carbon steel, 1.1% max. carbon, diameters 9/16 to 1 in.
Lead	Commercial purity virgin lead, B-29
Brass pipe	SB-43
Brass fittings	SB-249

10.2.3 SHIELDING DESIGN. The basis for the shielding design of the thermal shield is to reduce the energy current from the reactor by a sufficient amount so that the concrete biological shield will not be subject to thermal stress cracking. Since the thermal shield is entirely contained within the bulk concrete shield, activation of the thermal shield material is not an important consideration. Conventional materials, steel, lead, and cooling water, are used in the thermal shield. The thickness of these materials must be sufficient to reduce the energy current density into the concrete to something less than 7.5×10^{11} Mev/cm²-sec, which is the energy current of 8 Mev gamma rays which would give a 50°F temperature rise in the concrete. The 50°F maximum temperature rise is set as a conservatively small temperature rise which the shield concrete will sustain without danger of cracking.

Table 10.2-2 summarizes the various radiation components which are incident on the thermal shield and which penetrate it to enter the concrete. Both number and energy current densities are given. Table 10.2-2 refers to the horizontal midplane of the reactor. At points above or below the midplane the incident radiation per unit area of shield surface is reduced. In particular, this tends to reduce the energy loading on the upper thermal shield.

Table 10.2-2 Shielding Summary Table For Thermal Shield, At Horizontal Midplane of Reactor

Source Component	Energy per Particle Mev	Number Current into Thermal Shield n/cm ² -sec	Energy Current into Thermal Shield Mev/cm ² -sec	Number Current into Concrete n/cm ² -sec	Energy Current into Concrete Mev/cm ² -sec
Group 1 Core Gamma	6	5.09×10^{10}	3.05×10^{11}	5.05×10^8	3.03×10^9
Group 2 Core Gamma	4	1.56×10^{12}	6.25×10^{12}	3.44×10^9	1.38×10^{10}
Group 3 Core Gamma	2	1.81×10^{12}	3.61×10^{12}	5.64×10^9	1.13×10^{10}
Group 4 Core Gamma	1	3.60×10^{10}	$\frac{3.60 \times 10^{10}}{1.02 \times 10^{13}}$	4.17×10^7	$\frac{4.17 \times 10^7}{2.82 \times 10^{10}}$
Total Core Gamma			1.02×10^{13}		2.82×10^{10}

Thermal & Epithermal Neutrons (E<1Mev)	7.5	5.4×10^{12}	4.05×10^{13}	2.32×10^9	1.74×10^{10}
Fast Neutrons (E>1 Mev)	8	6.5×10^9	5.2×10^{10}	1×10^9	8×10^9
Capture Gammas from Thermal Shield	8	--	$\frac{- -}{4.06 \times 10^{13}}$	6.89×10^9	$\frac{5.50 \times 10^{10}}{8.04 \times 10^{10}}$
Total "Neutron" Energy			4.06×10^{13}		8.04×10^{10}

TOTAL			5.08×10^{13}		1.09×10^{11}

Detailed consideration of the attenuation of the various radiation components in the upper and lower thermal shield yields the heating rate curves shown in Figure 10.5. In the calculation of the heating rate in the lower thermal shield, the geometry of the various slabs of material in the shield wall was included in the calculation. The results show the discontinuities in heating rate at the several slab boundaries. In the calculation for the upper thermal shield heating rate, the shield was assumed to be a homogeneous iron-water mixture (61.5% iron)³. The heating rate is then a smooth curve, and is plotted in terms of watts/cm² heating in equivalent solid steel. The upper shield equivalent "solid thickness" is 31.24 cm, which is 61.5% of the 20 in. actual thickness.

10.2.4 THERMAL AND MECHANICAL DESIGN. The mechanical arrangement of the thermal shield is so strongly influenced by the radiation heating of shield members and their subsequent temperature expansion that the thermal and mechanical design of the shield must be considered jointly. In the lower thermal shield, which sustains the highest heating rates, several carbon steel plates line the inside of the shield to absorb the greatest radiation heating impact and attenuate the thermal neutron flux so that more remote sections of the shield are not unduly heated. The radiation heating load on these inner plates is so great that the thickness which can be used in a single plate is limited, and there is the further requirement that rapidly flowing water be in contact with these plates during all high power operation. The inner shell of the thermal shield is 3/4 in. thick (plate A). Proceeding outward, it is followed by a 1/2 in. cooling water annulus, called the inner annulus, with upward water flow. Next there is a 1/4 in. thick carbon steel plate which serves as a flow divider (plate B). Next there is a 3/4 in. thick water annulus, with downward flow, called the outer annulus. Next is a 1 in. thick carbon steel plate (plate C). Then follows the 4-3/4 in. thick lead layer, referred to as region D. Lastly there is a 1 in. thick carbon steel plate which is the outer wall of the thermal shield (plate E).

The coolant water is admitted to the lower thermal shield through an 8 in. inlet pipe which penetrates the bottom head to the inner annulus. Water flow is upward in the inner annulus to the top of the shield, reversing around the flow divider plate and coming downward in the outer annulus. In addition there is a set of cooling coils in the lead layer, set out against plate E. These coils form a network in the outer portion of the lead layer to keep down the temperature of the outer wall of the shield. The coils also loop around beam tube liners to reduce temperatures in these highly heated members. The normal water flow rate in the lower shield is 700 gpm, which gives velocities of 1.56 ft/sec in the inner annulus and 1.01 ft/sec in the outer annulus. The normal water flow rate in the cooling tubes is 30 gpm, which gives a velocity of 0.45 ft/sec.

The energy to be removed in the cooling water annuli and tubes may be calculated from the heating rate curves of the previous section. The heat flux from plate A at 40 MW reactor power is 13,000 Btu/hr-ft². Plate B, the flow divider plate, has a total heat flux of 2900 Btu/hr-ft² to be dissipated between the inner and outer cooling annuli. Plate C passes a heat flux of 12,000 Btu/hr-ft² to the outer cooling water annulus. Part of this energy is generated in plate C itself, and part is conducted through plate C from the lead layer. The coolant tubes at the outer surface of the lead layer have a heat flux of 7300 Btu/hr-ft², arising in part from the lead layer energy generation and in part from the radiation heating in plate E. These data, together with more detailed information on the thermal design of the lower shield may be found in Table 10.2-3. This table lists a variety of thermal design parameters and temperatures for various flow rates through the lower thermal shield at

Table 10.2-3 Effect of Coolant Flow on Critical Metal Temperatures in Lower Thermal Shield

Parameter	Unit	Inner Annulus				Outer Annulus				Coolant Tubes		
		Plate A Metal Temperatures				Plate C Metal Temperatures				Plate E Metal Temperatures		
Flow Rate	gpm	600	750	1,000	1,250	600	750	1,000	1,250	27	54	108
Flow Area Annulus	ft ²	1	1	1	1	1.53	1.53	1.53	1.53	.145	.145	.145
Flow Velocity	ft/sec	1.33	1.67	2.23	2.79	0.87	1.09	1.45	1.81	.408	.815	1.63
Reynolds Number	- -	18,200	22,800	30,400	38,000	17,850	22,250	29,700	37,000	4,600	9,200	18,400
Pressure Loss	psi		5.2	9.3	14.5	Included in Plate A				0.7	2.5	10
Film Coefficient	Btu/hr/ft ² °F.	440	526	665	800	287	344	430	515	178	310	540
Scale Coefficient	Btu/hr/ft ² °F.	1,000	1,000	1,000	1,000	1,000	1,000	1,000	1,000	1,000	1,000	1,000
Composite Coefficient	Btu/hr/ft ² °F.	305	345	400	445	223	256	300	340	150	247	350
Heat Flux	Btu/hr/ft ²	13,000	13,000	13,000	13,000	12,000	12,000	12,000	12,000	7,300	7,300	7,300
Film Temperature Difference	°F.	29.6	24.7	19.6	16.2	42	34.9	27.9	23.3	41	23.5	13.5
Scale Temperature Difference	°F.	13	13	13	13	12	12	12	12	7.3	7.3	7.3
Composite Temperature Difference	°F.	42.6	37.7	32.6	29.2	54	46.9	39.9	35.3	48.3	30.8	20.8
Coolant Temperature	°F.	110	110	110	110	115	115	115	115	125	120	110
Metal Temperature - Coolest	°F.	152.6	147.7	142.6	139.2	169	161.9	154.9	150.3	173.3	150.8	130.8
Metal Temperature Difference	°F.	14.2	14.2	14.2	14.2	23.4	23.4	23.4	23.4	2	2	2
Metal Temperature - Hottest	°F.	166.8	161.9	156.8	153.4	192.4	185.3	178.3	173.7	175.3	152.8	132.8
Metal Temperature - Average	°F.	162.0	157.0	152.0	149.0	185	178	171	166	174.3	151.8	131.8
Temperature Rise in Coolant	°F.	7.0	5.6	4.2	3.4	6.5	5.2	3.85	3.1	32	16.0	8.0

40 MW reactor power.

The cooling water to the thermal shield, both upper and lower, is supplied from a process system consisting of pumps, heat exchangers, and appropriate valves and instrumentation located on the equipment level of the building. This process system is discussed in detail in Section 7.9.

The temperature distributions in the lower thermal shield steel plates for both the normal coolant circulation condition and for the case of loss of flow in the cooling coils are shown in Figure 10.6. The temperatures of the various steel plates in the bottom head of the lower thermal shield are shown in Figure 10.7.

The normal coolant flow through the upper thermal shield assembly is 100 gpm, with a pressure drop of 3.3 psi. A maximum flow rate of 200 gpm has been set in the design. Cooling water passes through the split-ring sections of the upper thermal shield in series through four quadrant sections of the split-rings. Coolant flow is supplied in parallel to the support ring of the upper shield assembly. Figure 10.7 shows the temperatures in the various plates of the upper thermal shield at 40 MW reactor power, with the standard 100 gpm coolant flow rate through the upper shield.

The mechanical design of the thermal shield is based on an internal water pressure in the coolant spaces of 16 psig, the weight of the upper thermal shield (30,000 lbs), the weight of the reactor vessel and contents (45,325 lbs), and the effect of the various temperature differences between the plates of the shield structure. Various transient loadings are also considered, in particular an earthquake loading amounting to an acceleration of 0.1 g in any horizontal direction, an internal pressure in the thermal shield cavity of 400 psig, and a dry scram shock load on the reactor vessel of 560,000 lbs. Various thermal transients in the thermal shield are also considered in the design.

In the lower thermal shield, it is found that the basic plate structure is an exceedingly strong one with regard to vertical weight loads from the reactor vessel and upper thermal shield. These loads contribute such small stresses to the lower thermal shield plates that they may be ignored. The internal water pressure gives rise to stresses which are of significant magnitude only at the upper ends of plates A and C. Here the plates have to be made thin to provide flexibility and the pressure stresses become appreciable. In the rest of the plate structure of the lower thermal shield, the water pressure is too low to cause significant stresses. The main loadings on the lower thermal shield which give rise to significant stresses are the differential expansions.

The various steel plate shells of the lower thermal shield, plates A, C, and E, are rigidly connected at the horizontal mid-plane of the assembly by the nine beam tube liners. These are thick-walled cylinders which connect plates A, C, and E and which are so short that there is relatively little flexibility between the plates in the plane of the beam tubes. With the rigid connection of the plate structures at the horizontal mid-plane, it is clear that temperature differences from plate to plate will lead to different deflections at the end points of the various shells. There are two sets of differential expansions which are of interest. The first of these is between plates A and C. The inner shell, plate A, is free below the plane of the beam tube liners, and differential expansions on the lower side of the mid-plane are of no consequence, i.e., they lead to no stresses. At the upper ends of plates A and C, however, there must be a connection which seals the water passages. To obtain the necessary flexibility between plates A and C in this direction the upper ends of A and C are thinned down so that deflection under the temperature

expansion may take place with acceptable stress levels.

The differential expansions between plates C and E are more difficult to deal with, and of greater magnitude, because the expansion of the lead layer between these plates must be taken into account. Again, plates C and E are rigidly connected in the plane of the beam tube liners. In the lower head, plates C and E are connected by the inlet and outlet nozzles for the coolant water. These nozzles are located as close to the center of the bottom head as possible, so that the horizontal radius of the bottom head is available as a deflecting arm to relieve differential thermal expansions. Because the horizontal arm is of substantial length, 30 to 45 in., the stresses in this region are not severe. Above the plane of the beam tube liners, however, plates C and E are rigidly connected with a relatively short horizontal arm for deflection. As a result the largest stresses in these plates appear in the deflecting member at the top of plate C. These stresses are of an acceptable magnitude for long term service so long as the average temperature differential between plates C and E is no greater than 50°F.

A diagram of the lower thermal shield wall and a table of the stresses for various loading conditions is shown in Figure 10.8. The two thermal stress conditions are for normal operating temperatures ("steady state thermal"), and for the case of loss of coolant in the coils ("cyclic thermal"), as shown in Figure 10.6. The normal thermal and pressure stresses are relatively low. The thermal stresses for the loss of coil coolant case are large enough to require a fatigue analysis. The fatigue analysis (10.1) showed that the allowable number of cycles is about 2×10^5 . Since the loss of coolant case should occur only a few times at most in the life of the shield, the safety factor is very large.

In the upper thermal shield the structure is again a very massive and strong one. The stresses which result from the load of the reactor vessel and the weight of the upper thermal shield itself are not large. Thermal stresses in the upper shield are negligible because the differential expansions are small and the structure is relatively unconstrained. A diagram of the upper shield and a table of the stresses at various points is given in Figure 10.9. The "bursting pressure" loading referred to in Figure 10.9 is a 400 psig loading on the underside of the structure. The greatest stress for the normal condition occurs in the series of bolts which connect the inner flange of the support ring to the split-ring sections. The stress in these bolts is 22,700 psi. The bolts are of carbon steel, ASTM A-7, with a tensile strength of 60 to 72,000 psi and a minimum yield point of 33,000 psi.

Several thermal transients were examined to see if they imposed loading conditions which were important in the mechanical design. The first of these is the temperature transient in the various steel plates of the shield in the event of a reactor scram from steady state operation at 40 MW. The cooling water flow in the thermal shield is assumed to remain at the normal rate. The items of interest here, as in the steady state thermal calculation, are the average temperature differences between plates A, C, and E of the lower thermal shield. For the scram case, the maximum temperature difference between plates A and C occurs 1.8 minutes after the shutdown and has a value of 25°F. For plates C and E it was found that the temperature differences during the transient condition would be less than the 50°F limit for normal flow conditions.

The second thermal transient is the case of loss of coolant flow in the thermal shield which then precipitates a reactor setback. The heat capacity of the thermal shield is large and it requires about 1.6 hours for the afterheat to raise plate A and the adjacent water channel to the saturation temperature

(252°F). There is a greater thermal inertia involved in the rise in temperature of plates C and E, since these are bonded to the lead layer. For these plates it requires some four hours for the inner plate temperature to reach water saturation (252°F). The severest temperature differential between plates A, C, and E occurs in this case about 1 hour after the coolant failure and resulting reactor shutdown, and amounts to about 50°F.

From these considerations one concludes that transient thermal conditions in the thermal shield will not overstress the shield plate structures. In the event of a loss of coolant flow to the thermal shield, as in the second transient, the reactor is shut down automatically. The afterheat which appears in the thermal shield brings the static cooling water in the shield to saturation temperature in something less than 2 hours. The further accumulation of afterheat in the shield will be relieved by boiling of the water. It is a relatively easy matter to add makeup water to the thermal shield head tank and thus to supply whatever steam losses have to be made up as time goes on.

10.2.5 RADIATION DAMAGE. In considering radiation damage to the thermal shield the only concern is with the fast neutron bombardment of the steel plate structure of the shield. As noted in Table 10.2-2, the fast neutron current entering the thermal shield surface at the horizontal mid-plane of the reactor is 6.5×10^9 neutrons/cm²-sec, for neutrons of 1 Mev energy or greater. The accumulated fast neutron dose to the inner surface of the thermal shield, plate A, is given in Table 10.2-4.

Table 10.2-4 Fast Neutron Dose to the Thermal Shield Inner Wall, Reactor Horizontal Mid-plane.

Years of Operation at 40 MW	Megawatt-years of Operation	Fast Dose to Inner Wall
1 yr.	36.2 MW-yr.	1.85×10^{17} n/cm ²
10	362	1.85×10^{18}
20	724	3.70×10^{18}
30	1086	5.55×10^{18}
40	1448	7.40×10^{18}
50	1810	9.25×10^{18}

In Table 10.2-4 one may read years of operation at 40 MW power, or total megawatt years of operation at some other power. Since the HFBR is to operate 19 days in each 21, the on-time is nominally 90.5% and the operating year has 2.85×10^7 seconds. The fast neutron doses in Table 10.2-4 are given for the horizontal mid-plane of the reactor, which is the area of maximum dose. At points above and below the mid-plane, the dose is reduced below the values in Table 10.2-4.

The area of concern in terms of radiation damage to the shield is the possibility of brittle fracture of the steel plates. The plate material, SA-201, is a mild steel with an expected initial nil-ductility transition temperature (NDT) of about -20°F. The operating temperature of the steel plate structure

ranges from about 125°F to 180°F in normal circumstances. The irradiation of the shield is thus at low temperature, and no self-annealing effects are to be expected to ameliorate the radiation damage.

The accumulation of data on SA-201 to date (10.3) shows that an increase in the NDT of 100°F may be expected for an integrated exposure of about 4×10^{18} n/cm². A rise in the NDT of 150°F should occur at about 6×10^{18} , and a 200° rise at 9×10^{18} n/cm². Comparing these numbers to Table 10.2-4, one would expect to find the NDT in the thermal shield inner wall to have risen to about 80°F in a little over 20 years and to 130°F in a little over 30 years.

In assessing the hazards associated with a possible brittle fracture of the inner steel plate of the thermal shield, it is useful to keep in mind the fact that such a fracture would have no effect on the hazard which the HFBR presents to the general public, or for that matter, to the experimenters in the HFBR building itself. A serious fracture of the water-retaining plates in the thermal shield would cause a long and expensive shutdown of the reactor while repairs were made. But no radioactive materials are released by a fracture of the thermal shield, nor is the reactor vessel affected in any way, nor is the shielding capacity of the thermal shield diminished thereby. We do not wish to minimize our concern for the radiation damage problem in the thermal shield plates, but that concern is one of losing research time during the repair period, and is not a concern based on reactor hazards.

10.2.6 MANUFACTURE, TESTING, AND INSPECTION. The lower thermal shield was manufactured by the O. G. Kelley Company of Boston, Massachusetts, and the upper thermal shield was manufactured by CPC Engineering, Sturbridge, Massachusetts. As noted in Section 10.2.2, all of the carbon steel plate from which the shield was constructed was inspected by ultrasonic methods for laminations and stringing. All welders who worked on the thermal shield were qualified according to the procedures of the ASME Pressure Vessel Code, Section IX. Welds, in general, had full joint penetration, with double groove butt joints used where possible. In particular, the important water-retaining shells, plates A and C of the lower shield were made from single pieces of steel plate with a single longitudinal weld seam positioned in an area of minimum stress. The entire length of all welds and weld repairs was radiographed as the geometry permitted. In places where radiography was unsatisfactory, dye penetrant checks of the welds were made.

The manufacturer of the lower thermal shield was required to pour the lead layer so that there were a minimum of voids and unbonded surface areas. The manufacturing specification called for a lead thickness sufficient to maintain a gamma ray attenuation of not less than 90% of that of the nominal layer thickness. After the lead was poured between plates C and E a gamma scan was made of the entire lead layer. Because of the large gamma attenuation in the lead, it was not feasible in these particular circumstances to make x-ray exposures. The inspection method utilized a gamma ray source and sensitive gamma counters. A point-by-point scan was made over the entire cylindrical surface of the lower thermal shield and a similar scan over the bottom lead layer. The shield was judged to meet the requirements of the specification with regard to voids without exception.

In addition to the question of possible voids in the lead layer another important aspect of the lead pour was the bonding of the lead to the steel plates. The manufacturing specification called for 85% of the total interface area on the inner plate side to be metallurgically bonded and 58% on the outer plate side. Maximum unbonded areas in a single region were limited to 36 in.² except for areas on the outer plate side which were affected by the presence of

the cooling coils. The lead bond to the beam port liners was required to amount to at least 50% of the over-all interface area.

These requirements on the lead-steel bond were to insure adequate heat transfer contact between the lead layer and the steel plates. The metallurgical bond was checked by an ultrasonic inspection. The ultrasonic inspection was carried out on all surfaces to which the lead should have been bonded which were available for the work. (The cooling tubes imbedded in the lead layer were obviously not available for such an inspection.) The results of the ultrasonic inspection showed that the general quality of the bonding was excellent, with better than 90% of the total interface area between lead and steel having a metallurgical bond established.

However, there was a region on plate C where the incoming lead had splashed against the wall and had removed the tinning layer. The unbonded area in this region exceeded the maximum area for a single region and repairs were required. The repair was carried out in such a manner as to insure adequate heat transfer contact between plate C and the lead layer in the unbonded region. No attempt to rebond the lead in this area was considered feasible. The repair technique was to install a series of screwed and soldered plugs which were imbedded in the lead layer and which were then welded to the steel plate. These were placed on a rectangular grid of 4 in. spacing across the entire unbonded area. Before the repair was carried out, sample blocks simulating the thermal shield wall and lead layer were prepared and prototype plugs installed in the test block. Test blocks and practice plugs were then sectioned and examined for bond and general quality of construction. The test results showed the plug method of repair to be a satisfactory solution to the problem.

The assembled upper and lower thermal shields were each given a standard hydrostatic test at 25 psig in accordance with paragraph UG-99 of the ASME Code, Section VIII. At various stages in the manufacturing process, as subsections were completed, freon leak tests were made to insure that the tested welds did not have leaks. Frequent inspections in the manufacturers' shops by representatives of the Lummus Co. and BNL were made, and during much of the manufacturing period inspectors were in continuous residence.

10.3 BIOLOGICAL SHIELD

10.3.1 LOCATION AND FUNCTION. The biological shield is located in the center of the HFBR building. It extends from a base level at elevation 110 ft-11 in. to the pile top at elevation 149 ft-0 in. The lateral extent of the biological shield is approximately to radius 12 ft-6 in. from the vertical building centerline. The biological shield and the internal structures, the thermal shield, reactor vessel, and various reactor pit shield members, are supported by the reinforced concrete structure of the experimental level floor in the central area of the building. This level, in turn, is supported by the reinforced concrete walls of the shielded equipment cells on the floor below. These walls transfer the load of the biological shield and attached members to the foundation mat in the central area of the building.

The primary function of the biological shield is to reduce the radiation from the reactor core and thermal shield to an acceptably low level for human biological tolerance in the experimental area, and to levels low enough to permit low background neutron beam experiments to be carried out in this area. The upper portion of the biological shield serves the additional shielding function of protecting personnel from the radiation from used fuel elements as these are transferred from the reactor vessel to the storage canal. The biological shield is pierced with assorted apertures to permit neutron beams to

emerge, and to permit the insertion of samples into the various irradiation thimbles in the reactor.

The biological shield structure serves the important additional function of being the central structural column in the reactor building. The upper floor of the building, the operations level, is supported near the periphery of the building by a series of reinforced concrete columns which extend down through the experimental level and the equipment level to the foundation mat. At the center of the building the operations level floor is supported by the biological shield column. Another structural function of the biological shield is to provide the base support for the inner end of the experimental area crane.

10.3.2 GEOMETRY, DIMENSIONS, AND MATERIALS. The biological shield is shown in elevation in Figure 10.1, and in plan in Figures 10.10, 10.11, and 10.12, taken at elevations 149 ft-0 in., 127 ft-0 in., and 118 ft-0 in., respectively. The two figures 10.13 and 10.14 show the construction of the top plug structure for the biological shield. In the experimental area the outer face of the shield is a nine-sided irregular polygon. There are nine beam tubes which allow neutron beams to emerge into the experimental area, and each beam tube passes through one face of the polygon at 90° to the face. The sides of the polygon are arranged so that the distance from the center of the shield is very nearly 12 ft.-5 in. on all sides. This gives an effective diameter to the biological shield in the experimental area of about 25 ft. The dimension across opposite corners of the irregular polygon will, of course, be somewhat greater than this nominal diameter.

At an elevation of 132 ft-2 in., the irregular polygon shape of the shield is cut back to a uniform circular shape of radius 11 ft-4 in. This provides a ledge upon which the rail for the experimental area traveling beam crane is placed. In the region of the operations level floor, the biological shield is heavily reinforced by a massive ring girder which connects to the radial floor beams of the operations level floor. The function of this girder is to transmit across the biological shield structure the moments which result from a possible uneven heavy loading of the operations level floor.

The pile top, which extends above the operations level floor, is a regular octagon with a total distance across flats of 25 ft-4 in. A set of steel brackets support a catwalk around the perimeter of the octagon at the elevation of the pile top, 149 ft-0 in. (The catwalk may be seen in the elevation drawings of the building, Figures 3.1 and 3.2)

Inside the biological shield a heavy ring girder 9 ft in diameter and 2 ft high is imbedded in the heavy concrete at elevation 110 ft-11 in. The top surface of the ring girder is at elevation 112 ft-11 in., and forms the mounting surface for the thermal shield. The cavity into which the thermal shield fits is defined by a steel plate cylinder called the cavity liner. This member is of 1 in. steel plate, welded at the bottom to the ring girder, and at the top to other steel plates of the shield. The inside diameter of the cylindrical pit formed by the cavity liner is 9 ft-2 in. The pit extends from elevation 112 ft-11 in. to elevation 124 ft-6 in. At the bottom of the pit, beneath the thermal shield, a 4 in. lead layer is poured within the ring girder. Cooling coils are imbedded in this lead layer. A rectangular steel casing passes through the concrete beneath the thermal shield to furnish an aperture for the thermal shield cooling water lines. These pass upward from the basement process area to the thermal shield. The casing is filled with carbon steel shot packed around the water lines. The shielding beneath the reactor is adequate for personnel access during shutdown periods.

There is a set of cooling coils attached to the cavity liner and imbedded in the heavy concrete. These cooling coils extend vertically to cover the region of the thermal shield. They are provided to control the temperature of the inner face of the heavy concrete.

As shown in Figure 10.12, the neutron beam tubes extend more or less radially outward from the thermal shield to the outside of the biological shield at an elevation close to that of core centerline, 117 ft-6 in. The shutter blocks, and the cases in which they move, extend above the beam tube liners to elevation 124 ft-6 in. A plan view of the shutter blocks and cases is shown in Figure 10.11. Recesses in the shield concrete above this level to elevation 129 ft-4 in. are filled with stationary shielding blocks above each shutter, and also above the reactor inlet and outlet nozzles. The drive shafts for the movable shutter blocks pass through these fixed shielding members, as do the cooling water pipes to the movable blocks. In several locations around the periphery of the shutter recesses it is necessary to install lead block shielding to compensate for the void created by the shutters.

At elevation 129 ft-4 in., the pit widens to a circular cross section with inside diameter of 15 ft-4 in. The drive motors for the movable shutter blocks are installed at elevation 129 ft-4 in. Just above the shutter drive motors, at elevation 130 ft-8 in., a floor is established completely across the pit with a sealing gasket to a flange on the reactor vessel. This continuous floor plate establishes a tight gas seal between the upper pit and lower pit areas. Its purpose is to separate the air atmosphere of the upper pit, which is accessible to personnel during reactor shutdown periods, from the CO₂ atmosphere of the intermediate and lower pit, which is not normally accessible to personnel. The CO₂ atmosphere region extends downward from elevation 130 ft-8 in. and includes the thermal shield cavity, the beam tubes and shutter cases out to the sealing gaskets on the experimental plugs, the gas space between the thermal shield and the cavity liner, the coolant line case filled with steel shot beneath the thermal shield, and the pipe tunnels for the reactor coolant lines extending down to sealing boots at elevation 107 ft-3 in. The structure which supports the seal floor at elevation 130 ft-8 in. is also a shield to protect personnel in the upper pit from shutdown radiation from the reactor. The area beneath the seal floor to elevation 129 ft-4 in., around the shutter drive motors, is filled with stacked lead blocks to supplement the shielding in that area.

A light steel floor structure is installed in the pit at elevation 137 ft-5 in. This floor is the working level for personnel engaged in fuel changing and experimental thimble maintenance operations. The reactor vessel head extends an inch or two through this floor and it is at this point that the vessel is opened for fuel handling. This upper room of the reactor pit contains a hoist on a swinging beam for handling the reactor top cover and other heavy members which may come out of the reactor. The side walls of the shield are pierced in this region by standard beam tube liners (closed with solid plugs) for future access to the upper pit for experimental purposes.

The reactor pit is sealed at the top by a plug structure which is shown in section in Figure 10.1, with details of the assembly in Figures 10.13 and 10.14. The top plug structure is 4 ft thick, composed of steel plate and heavy concrete. It is bolted to the stationary shield structure. A central aperture of 22-5/8 in. diameter is provided for fuel handling purposes. The basic structural piece in the top plug is the cartwheel shown in Figure 10.14. This member is 16 ft-0 in. in diameter, 20 in. thick, and is the support for the several shielding plugs which compose the top shield. Individual plug segments, nine in number, are set into the cartwheel and individually bolted in

place. In addition to the central hole for fuel handling, the top plug is pierced by several periscope ports, holes for the vertical irradiation thimbles V-13, 14, 15, and 16, and by a manhole plug. In normal operation all of these penetrations of the top plug are sealed. The central hole is covered by the fuel handling shutters, and the other holes by solid plugs or other shields.

The materials of the biological shield structure are steel plates, lead, and heavy concrete. The steel plate material is ASTM A-7 for all of the plate used in the various shield plugs. The shield face plates, both on the outer face and on the lining of the pit, are ASTM A-201, grade B. These are both mild steels with minimum yield points in the range 30 to 33,000 psi and tensile strengths from 60 to 72,000 psi. The retaining bolts for the top shield plug segments and for the cartwheel section of the top plug are of AISI C-1035. This is a steel with a yield point of 45,000 psi, an 85,000 psi tensile strength, and an elongation of 25%. The lead shield components in the biological shield are bricks of standard commercial purity lead.

The heavy concrete is a mixture of Portland Cement, limonite iron ore, steel punchings, water, and a liquid plastiment. The mix composition used in the biological shield concrete is as follows:

Cement	840 lbs/cu. yd.
Limonite ore	1856 " " "
Steel punchings	4690 " " "
Water	298 " " "
Plastiment	45 fl. oz./cu. yd.

The mixture listed above gives a wet mix weight of 7684 lbs/cu. yd. or 285 lbs/ft³. The calculated dry density is 279 lbs/ft³. Field tests on the mixture, done by weighing test cylinders, gave a dry density of 276 lbs/ft³. As-sorted properties of the heavy concrete are listed below in Table 10.3-1. It must be recognized that the constitution of the heavy concrete mix undergoes minor variations from one pour to another. Further, the composition of the iron ore aggregate changes from one lot to another, giving slightly different properties for successive batches of what is nominally the same concrete.

10.3.3 SHIELDING AND THERMAL DESIGN OF THE BULK SHIELD. The radiation sources for the shielding design of the bulk shield are the gamma ray and neutron currents given in Table 10.2-2. The values in Table 10.2-2 are for the horizontal mid-plane of the reactor, which is the level of maximum radiation current into the concrete. The bulk shield outside the cavity liner plate has a variable thickness, since the cavity liner is a circular section and the outer surface of the bulk shield is an irregular polygon. The minimum bulk shield thickness, measured along perpendiculars to the outer faces of the shield, is 93 in. or 236 cm. The outer 1 in. of this thickness is the steel face plate of the biological shield. Considering the penetration of fast neutrons and gamma rays through this minimum bulk shield thickness at the reactor mid-plane, the expected current densities at the shield surface are 1×10^{-2} n/cm²-sec for fast neutrons and 6 Mev/cm²-sec for gamma rays. The gamma ray current corresponds to a dose rate of 1×10^{-2} mr/hr. At points on the outer shield surface above or below the mid-plane of the reactor the radiation levels should be lower than these, as will be the case toward the corners of the shield polygon.

The surface dose rates quoted above for the unperturbed bulk shield are

Table 10.3-1 Properties of BNL Iron-Limonite Concrete

Density	276 lbs/ft ³
Compressive Strength	5000 psi
Young's Modulus	3.5 x 10 ⁶ psi
Coefficient of Expansion	6.8 x 10 ⁻⁶ in/in-°F
Specific Heat	0.164 BTU/lb-°F
Conductivity	2.06 BTU/hr-ft-°F
Diffusivity	0.0455 ft ² /hr
Fermi Age to Thermal	100 cm ²
Fast Neutron Removal "Cross Section"	0.135 cm ⁻¹
Gamma Absorption Coefficients	
3 Mev	0.150 cm ⁻¹
4 "	0.138 "
5 "	0.128 "
6 "	0.1235 "
7.6 "	0.1178 "

very low. However, they bear little relation to the dose rates to be expected outside the shield of the actual reactor. The reason for this is twofold. First, the bulk shield is pierced in the vertical direction by a number of air vents, pressure relief vents, and coolant water pipes. These lines may be seen in the horizontal section of the biological shield at elevation 118 ft-0 in., shown in Figure 10.12. Each of these imbedded pipes or vent lines is a void area in the heavy concrete shielding which inevitably causes a local high intensity spot for both fast neutrons and gamma rays. The presence of these members may increase the local radiation level at points on the shield face by factors of as much as 30.

The second reason is that the bulk shield is pierced by the nine neutron beam ports. These ports extend completely through the concrete portion of the shield and open through apertures in the thermal shield to the beam tubes in the reactor vessel itself. These ports are obviously a source of very intense streaming of neutrons and gamma rays from the active core region to the outer reaches of the shield. The degree to which this radiation streaming is reduced by limiting the aperture for the emerging neutron beam and by collimating to a small angle of divergence varies from experiment to experiment. For all experiments, a substantial bulk of shielding material is required at the face of the biological shield to absorb neutrons and gamma rays which are scattered in the various elements of the experimental apparatus. It is clearly not practicable to attempt to consider all of the possible experimental plug and shield geometries to deduce "standard" values for neutron and gamma ray dose rates in the experimental area. The general levels to be expected from the experimental apparatus will approach biological tolerance levels for both fast neutrons and gamma rays. The control of these radiation levels on the experimental level will be a matter of continuing health physics survey work. It is quite possible that certain experiments will scatter radiation to levels above biological tolerances and that areas of the experimental level will be classed as "limited access" for the duration of such experiments.

There are two more areas of interest in the shielding design. The first of these concerns a large gamma ray source from neutron capture at the juncture of the spherical section of the reactor vessel with the cylindrical neck. The capture gamma source in this area is the stainless steel structure in the transition plate and in the control rod followers and blades. This source is high enough in the reactor vessel so that a substantial cone emerges from the central hole in the upper thermal shield. At the several locations around the perimeter of the pit where the shutter block voids appear and where the inlet and outlet heavy water lines pass, the gamma field is of sufficient intensity to require backup lead shielding at the inner surface of the heavy concrete. These lead shields are shown in the elevation drawing, Figure 10.1, and in the horizontal section at elevation 127 ft-0 in., Figure 10.11.

The second special shield which is required is a lead shield tapering from 8 in. thickness to 2 in. thickness in the region of the fuel chute at elevation 112 ft-0 in. to elevation 120 ft-0 in. In this region the fuel chute passes close to the port box on beam tube H-2. In the course of the fuel changing operations the irradiated fuel elements are passed down the chute to the canal. Normally they will pass the H-2 port box region in a short time, and the radiation pulse which results from their passing will yield a very small integrated gamma ray dose. However, the fuel chute shielding must be designed with the possibility of a jammed fuel element in mind. Since such an event is regarded as quite unlikely, the dose rate from a jammed fuel element at the worst place by the H-2 port box has been set at 1 r/hr. The lead shadow shield around the chute accomplishes this reduction.

The heating rates in the concrete at various distances from the inner surface are shown in Table 10.3-2. The temperature distribution in the bulk concrete shield is calculated from these data, with the assumption of a fixed temperature at the inner surface. The temperature rises a total increment of 7.5°F, reaching essentially this value at a point 20 in. from the inner surface of the concrete. The assumption of a fixed temperature at the inner surface is based on the cooling coil array which is attached to the cavity liner plate and imbedded in the heavy concrete at that point. The inner face of the concrete is held at 110°F by the cooling coils, so that the maximum concrete temperature is 117.5°F at about 20 in. into the concrete. The average temperature is 103°F, and the outer face is at 90°F. The inner portion of the concrete will be in compression from this temperature variation, with the compressive stress at the inner face being 174 psi. The maximum compression occurs at a distance of about 8 in. into the concrete, where the stress is 474 psi. About 54 in. from the inner surface of the concrete the thermal stress changes from compression to tension and rises to a maximum value of the outer surface of the shield of 302 psi. The compressive stresses are very small compared to the compressive strength of the concrete. The tension stress is a more difficult matter for concrete, but is below the normally accepted value of 1/10 of the compressive strength. Further, the outer face plates of the biological shield form a substantial reinforcement which is not included in any of these calculations. If the water flow in the cooling coils is cut off, the temperature of the cavity liner and inner concrete surface rise to 135°F and the point of maximum compressive stress moves to the inner surface. The compressive stress there is about 550 psi. The outer surface of the concrete has a temperature of 93°F, and the maximum tension stress (which is at the outer surface) is about 460 psi. The transition from compression to tension occurs, in this case, 47 in. from the inner surface (10.4).

10.3.4 SHIELDING IN THE UPPER PIT. This section deals with shielding design in the upper pit from two aspects. The first of these is the shielding of the upper pit room from the active core during operation at power and at

Table 10.3-2 Heating Rates at Various Distances Into The Concrete Biological Shield, at the Reactor Mid-plane

Source Component	Particle Energy	Energy Deposition at:			
		0 in.	6 in.	12 in.	18 in.
	Mev		Mev/cm ³ -sec		
Capture Gammas, Inner Steel Plate of Thermal Shield	8	2.51x10 ⁸	5.53x10 ⁷	1.18x10 ⁷	2.70x10 ⁶
Capture Gammas, Lead Layer of Thermal Shield	8	1.66 "	1.93 "	.59 "	5.15 "
Capture Gammas, Outer Steel Plates of Thermal Shield	8	52.3 "	47.7 "	8.40 "	16.3 "
Capture Gammas in Concrete	8	16.3 "	87.6 "	18.6 "	36.0 "
Group I Core Gamma	6	3.03 "	8.10 "	1.63 "	2.91 "
Group II Core Gamma	4	13.2 "	31.1 "	5.45 "	8.50 "
Group III Core Gamma	2	10.3 "	14.2 "	1.98 "	2.80 "
Fast Neutron Slowing Down	3	8.81 "	12.1 "	1.68 "	2.29 "
TOTALS:		1.08x10 ¹⁰	2.08x10 ⁹	3.95x10 ⁸	7.67x10 ⁷

shutdown. The other aspect deals with radiation levels outside the shield walls of the upper pit during the passage of irradiated fuel elements to the canal.

We deal first with the shielding of the upper pit rooms from the active core. The upper pit is, of course, not available to personnel while the reactor is operating. However, some reasonable time after shutdown, operating personnel must be able to enter the upper pit room to handle fuel, service the experimental facilities, and do maintenance work on the control rod drive mechanisms.

The various shielding layers which protect the upper pit space from the active core are first the heavy water in the cylindrical portion of the reactor vessel, next the thermal shield cover, next the lower service floor shielding, and finally a 4 in. thick collar of lead around the neck of the reactor vessel above the pit seal floor. The floor of the upper pit room is slightly more than 12 ft above the top of the active core.

The thermal shield cover is 20 in. thick, and is a 60% steel, 40% water shield. The lower service floor is 17 in. thick, and contains 2 in. of steel plate, 6 in. of lead, 1/4 in. of boral, and the balance heavy concrete. To reduce activation gamma sources in the intermediate pit region the reactor vessel is wrapped with boral sheet above the vessel support bracket level. The inlet and outlet nozzles are also wrapped with boral, as is the portion of the reactor vessel above the pit seal floor.

To reduce the activation gamma source from stainless steel structures inside the reactor vessel neck, it was necessary to introduce thermal neutron absorbers into the heavy water column. These members, called neutron shields, are mounted on the control rod drive mechanisms inside the reactor vessel (see Section 9.7.4.4). The shields extend from approximately elevation 124 ft-0 in. to elevation 130 ft-6 in. The shields greatly reduce the relaxation length for thermal neutrons in the reactor vessel neck, and thereby reduce the activation gamma source from the control rod structures high in the vessel. Figure 10.15 shows radial thermal flux plots in the neck of the vessel; the left plot at elevation 180 cm above core center and the right plot at elevation 300 cm above core center. The effect of the neutron shield absorbers is seen in the right-hand plot in Figure 10.15. In Figure 10.16 the axial variation of the thermal flux, due to core neutrons in the left plot, and to core plus photoneutrons in the right plot is shown. These axial graphs are taken at radius 35 cm in the vessel neck. The neutron shields are so effective in reducing the thermal flux from the core that the photoneutrons dominate at the elevation of the lower service floor and above.

Table 10.3-3 lists the fluxes and dose rates from various radiation sources, both at 40 MW operation and 24 hours after shutdown. These radiation levels are for the control rod drive room in the upper pit, i.e., the lower level to which personnel have access. For the purpose of establishing the cobalt activation gamma doses, 10 years of operation at 40 MW power has been assumed. An important qualification for Table 10.3-3 is that all penetrations through the lower seal floor and the upper thermal shield are assumed to be plugged with shielding equivalent to the floor section and the thermal shield section. These penetrations for experimental facilities and various reactor piping cannot be plugged in this simple fashion, however. Shielding collars will have to be built up around the penetrations. The plugging of inevitable crevices in the shielding layer at the lower service floor level will be carried out during the startup period for the reactor, when operation at low power level will permit careful gamma and neutron surveys to be made in the upper pit region.

Another feature of the radiation level in the upper pit after shutdown is the activation gamma ray dose from sodium-24 in the primary heavy water. The concentration of Na-24 to be expected in the heavy water at various times after shutdown and for various demineralizer flow rates is shown in the left-hand graph of Figure 10.17. In the right-hand graph of Figure 10.17 are the results of a dose rate calculation for the Na-24 gammas at the vessel top flange, again for various times after shutdown and various demineralizer flow rates.

We turn now to the second aspect of the shielding problem in the upper pit, that of the dose rates outside the shield walls to be expected in fuel handling operations. Individual fuel elements are lifted from the reactor vessel, placed in the top of the fuel chute, and are released to slide into the canal in the equipment area. This transfer is carried out by remote handling equipment. The thinnest portion of the pit shield wall is found above the crane rail at elevation 132 ft-2 in. The wall thickness is 43 in., including 2 in. of mild steel plate. The most active fuel elements to be removed from the core will give rise to a gamma ray dose rate just outside this wall section of 4 mr/hr. The dose rate through the 48 in. thick top plug structure will be less than 1 mr/hr. The steel and lead shutters which cover the central fuel handling hole through the top plug should limit the dose through that aperture to about 1 mr/hr. Crevices between plug sections, or between periscope tubes and liners, etc., may lead to local "hot spots" on the top plug. These areas will be detected during shielding surveys in the startup period and will be covered with appropriate shielding.

Table 10.3-3 Radiation Levels in the Control Rod Drive Room In the Upper Pit

Radiation Source	40 MW Operation		24 hrs. After Shutdown	
	Flux ₂ n/cm ² -sec	Dose Rate rem/hr	Flux ₂ n/cm ² -sec	Dose Rate mrem/hr
FAST NEUTRONS:				
Core Fission	5x10 ⁵	125	9	2.3
Photoneutrons	1.4x10 ⁵	35	2.4	.6
THERMAL NEUTRONS:				
Concrete in floor	<10 ⁴	nil	nil	nil
Concrete walls	1.8x10 ⁵	1.35	3	.023
CORE GAMMA RAYS:				
Prompt fission & capture		nil		nil
Fission product decay		nil		.11
CAPTURE GAMMA RAYS:				
Control rod guides (Fe)	2x10 ⁷	186	350	3
Iron in floor		nil		nil
Iron in concrete walls	6x10 ⁵	5	10	.1
Iron pit liner plate	3x10 ⁶	30	50	.5
Upper thermal shield (Fe)	3x10 ⁵	3	5	.05
Reactor vessel neck (Al)	1.9x10 ⁵	1.7	3.2	.03
ACTIVATION GAMMA RAYS:				
Control rod guides (Co)		nil	1200	2.8
Cobalt in floor		nil		nil
Cobalt in concrete walls		nil	900	2.1
Cobalt in pit liner plate		nil	2100	4.8
Upper thermal shield (Co)		nil	3	nil
Equipment in room (Co)		nil		7x10 ⁻⁵ per cm ³ steel

10.4 PRIMARY EQUIPMENT CELLS

10.4.1 DESCRIPTION. Most of the primary heavy water process equipment is housed in the shielded cells on the equipment level. There are three cells, grouped together to form a rectangular concrete structure 35 ft-0 in. wide by 61 ft-6 in. long, and extending from the equipment level floor to the experimental level floor. The cell block is approximately centered in the reactor building. The arrangement of these cells may be seen in Figure 3.3, and in elevation in Figures 3.1 and 3.2.

The northern cell is 25 ft-0 in. by 10 ft-0 in. by 14 ft-3 in. high (inside dimensions). It contains the primary coolant down-comer from the reactor vessel, the main flow meter for the primary system, the main pump suction valves HC-102A and B, the shutdown cooler, EA-103, the helium eductor pump GA-105, and assorted primary and thermal shield coolant system piping. The other two cells have interior dimensions of 41 ft-6 in. by 10 ft-0 in. by 14 ft-3 in. ceiling height. Each contains a main pump, GA-101, a main pump check valve, a main heat exchanger, EA-101, a primary loop throttle valve, HCV-101, an air conditioning unit, and assorted primary and secondary system piping. The eastern cell is for the "A" loop components and the western cell for the "B" loop components.

The floors of the three cells are formed by the equipment level foundation mat, which has a thickness of 5 ft and is of heavily reinforced concrete. The cell ceilings are of reinforced concrete, 6 ft-3 in. thick except in the region of the biological shield, where the ceiling thickness is 3 ft-8 in. of reinforced structural concrete and 2 ft-7 in. of heavy concrete. Walls between and around the cells are 5 ft thick, of reinforced structural concrete. The walls of the shielded cells are not only for the purpose of shielding the primary system but also are structural support members for the portion of the experimental level floor above the cells. In particular, the interior walls separating the cells are the primary support columns for the biological shield structure.

Access to the north cell is through a 4 ft wide by 6 ft high opening on the west end of the cell. Like all access doors to the shielded cells, this one has a 12 in. high sill. Shielding coverage for the access door is provided by a movable stacked block shield outside the wall of the cell. The movable shielding door is 5 ft thick, 9 ft long, and 9 ft high. It is composed of solid cement blocks stacked on a base carriage which rolls on rails set in the floor. The atmospheric seal for the north cell access door is provided by a steel plate door on the inner side of the shield wall, with a soft rubber gasket compressed between the door and a fixed steel frame set in the shield wall concrete.

Access to the "A" and "B" loop cells is principally through man-entry doors on the east and west sides of the cell block. There are large access doors at these points cast in the basic shield walls. These entry ways give a free width of 5 ft-8 in. and a free height of 12 ft-9 in. Again there is a 12 in. sill at the doorways. This large access aperture is for the replacement of primary pump components. In normal operating situations, the large entry is blocked by stepped shielding blocks to leave a personnel access door of dimensions 2 ft-6 in. width by 6 ft-5 in. height. Movable shield block sections, mounted on wheeled carriages, allow easy removal of the shield door for entry. The atmospheric seal for these entries to the "A" and "B" loop cells is provided by steel plates which are bolted down on soft rubber gaskets, and which cover the full height and width of the aperture in the shield wall. Set into these steel plates are man-doors with clear openings of 2 ft-6 in. by 6 ft. The man-doors, in turn, also close on soft rubber gaskets.

At the south ends of the "A" and "B" loop cells, there is no fixed shield wall except for a 12 in. high sill. The open ends of the cells permit the removal of the main heat exchangers. Shielding coverage for these open ends is provided by stacked walls of ordinary solid concrete block, with the wall dimensions being 5 ft thickness, 30 ft length, and 15 ft height. The atmospheric seal at these open ends of the loop cells is provided by steel plates which bolt down on soft rubber gaskets.

The sills on all access doors to the shielded cells are to retain heavy water leakage. Should a major break occur in the primary system, the heavy water leaking out would accumulate on the floors of the three cells. The sills are of sufficient height so that the entire primary inventory could be dumped into the shielded cells without overflowing the sills. Each of the three cells is drained by a floor drain to the FA-101 sump, a covered storage tank pit used as the ultimate drain collection point for all heavy water spillage.

Emergency pressure vent pipes from the lower pit space in the biological shield terminate in the loop cells, one 12 in. pipe in each cell. Since these pipes are open-ended at the top to the CO₂ atmosphere of the lower pit, seals

must be provided. These seals are rupture discs bolted to flanges at the lower ends of the pipes in the loop cells. The rupture discs are aluminum, rated at 6 psi differential pressure.

The building air exhaust system is so arranged as to take suction on the north shielded cell, drawing about 400 cfm from that space. In turn, the "A" and "B" loop cells exhaust into the north cell, with air coming into the loop cells from ventilation louvers in the steel plates at the south ends of the cells. The reactor upper pit room also exhausts to the north shielded cell through a 12 in. exhaust line. The pressure of the shielded cell atmosphere is kept at a point somewhat below the surrounding air pressure in the equipment level. Thus, air leakage is inward to the shielded cells rather than outward to the equipment level atmosphere.

Some heat removal is necessary in the shielded cell block, particularly in the "A" and "B" loop cells where the two main pumps generate a considerable heat load. The air conditioning units IB-310A and B are located in the "A" and "B" loop cells. These units circulate 3,000 cfm air each and have a total heat removal capacity of 235,000 Btu/hr. The temperature and humidity conditions for the shielded cell spaces are 95°F, and a maximum relative humidity of 27%.

10.4.2 SHIELDING DESIGN. While the reactor is operating at any appreciable power, the principal radiation source within the shielded cells is the N-16 gamma activity in the primary coolant. Estimates of the N-16 concentration at the vessel outlet which have been made vary from 40 to 100 $\mu\text{C}/\text{cm}^3$. For dose rate calculations and shielding work it is appropriate to use values on the conservative side, and for this section we assume the N-16 concentration at 40 MW reactor power to be 100 $\mu\text{C}/\text{cm}^3$ of heavy water. The N-16 dose rate in the shielded cells at full power then runs from 100 to 200 r/hr. The dose rate from the N-16 gamma rays outside the 5 ft thick concrete shielding walls of the cells is 2 to 4 mr/hr at the outer surface of the wall.

A few photoneutrons are produced from the interaction of the N-16 gamma rays on deuterium in the primary coolant. The flux level of these neutrons is too small to be of concern in terms either of the dose rate to materials or in activation of components in the shielded cells. Residual activities in the primary water itself far exceed any activation from photoneutrons. Similarly, the photoneutron flux outside the shielded cell walls is essentially zero.

During shutdown periods, the principal activity in the primary coolant will be Na-24. A graph of the concentration of Na-24 in the primary water as a function of time after shutdown, and for various demineralizer flow rates, is given in Figure 10.17. Dose rates from the Na-24 gamma rays are given in Figure 10.17 for the geometry of the upper end of the reactor vessel. Dose rates in the shielded cell rooms from Na-24 will be of the same magnitude as those at the vessel top, with the rates increased in some local areas by a factor of as much as 4 due to less favorable geometry. The general Na-24 dose rate level in the shielded cells will range from 3 to 10 mr/hr at 12 to 16 hours after shutdown. Should the sodium plate out selectively on one or another of the process components, so that local source concentrations build up above those in the bulk water, dose rates will rise accordingly. The dose rates outside the shielded cells at shutdown are essentially zero, since the 5 ft thick concrete walls provide an attenuation of about 1×10^{-5} for the Na-24 gamma rays.

10.5 MISCELLANEOUS SHIELDING

10.5.1 D₂O STORAGE TANK FA-101. The collection point for the heavy water floor drains is the FA-101 pit. This pit is below the level of the equipment

area floor, close to the perimeter wall (see Figure 3.3). The pit also contains the storage tank FA-101, an alloy steel 6,000 gal capacity tank. The pit is covered by a reinforced concrete slab 12 in. thick. The pit atmosphere is air, and is exhausted to the building filter bank. A slightly negative pressure is maintained in the pit so that air leakage is inward. The activity of heavy water which is collected in the FA-101 pit is not expected to be great in any normal circumstances. The 12 in. thick concrete cover is, therefore, expected to provide sufficient shielding for this area. Should fluids of greater than expected activity collect in the pit, its location by the perimeter wall makes possible the addition of more material on the cover and of shadow shield walls.

10.5.2 D₂O PURIFICATION SYSTEM. The three filter vessels and the four ion exchange vessels are contained in steel-lined pits sunk below the level of the equipment area floor. Each of these seven vessels is contained in an individual pit. The tops of the various vessels in the purification system are several feet below floor level, and are covered by reinforced concrete slabs. Again, the geometry itself gives a substantial shielding factor in all but the upward direction, and the arrangement of the beds is such as to permit easy addition of shielding blocks on the top of the cover slab. The pits, and trenches for connecting piping, have an air atmosphere which is exhausted to the building exhaust system so that the pit and trench volume is at a slightly negative pressure.

10.5.3 OPERATIONS LEVEL D₂O PROCESS AREA. All of the process components of the experimental heavy water system are situated in the operations area, as shown in Figure 3.6. In addition to these components, the second heavy water storage tank, FA-102, and the helium gas system for the reactor vessel are also located here. The area in which these process components are located is surrounded by a concrete sill to retain any fluid leakage, and a floor drain connects to the FA-101 collection pit. Since radioactivity must be expected in the piping and process units in this area, some shielding provisions are necessary. It is planned to stack ordinary solid concrete blocks around this process area, to form a shadow shield as required. Some additional local shielding above individual components may be required to reduce the radiation scattered over the shadow shield wall by the atmosphere.

10.5.4 D₂O SHUTDOWN PUMPS. The D₂O shutdown pumps, GA-102A and B are located in the equipment area at the north side of the shielded cell block. These pumps are outside the shielded cell (for easier maintenance while the reactor is operating) and require local shielding to reduce the radiation level from N-16 in the primary coolant. Each pump is surrounded by a concrete curb to retain liquid spillage, and is shielded by a lead block wall and lead and steel top cover. The design dose rate level at contact outside the shield walls around these pumps is 7 to 10 mr/hr. The electric motors for the shutdown pumps, as well as the operating wheels for several valves in the system, come through the top shielding layer so that they are accessible while the reactor is operating. The two pumps are individually shielded, so that one may operate while the other is accessible for maintenance work. The volume within the shield walls and covers around the pumps has an air atmosphere, exhausted to the north shielded cell.

10.5.5 AUXILIARY PROCESS SYSTEM SHIELDING. Several other auxiliary process systems need shielding to greater or lesser degree. These are the thermal shield coolant system, the biological shield coolant system, and the canal process equipment. All of these process systems are located in the equipment area at the north end of the shielded cell block. No permanent shield structures have been constructed for these components. As activity levels build up in the

various systems, shadow shields of ordinary concrete blocks will be constructed to reduce the local dose rates to an acceptable level.

The cement blocks to be used in these shadow shields, and in many other places for similar purposes throughout the facility, are commercially available blocks of dimensions 3-5/8 in. by 7-5/8 in. by 15-5/8 in. The block volume is .250 ft³. The blocks typically weigh 30 lbs, yielding a density for stacked shielding of this kind of 120 lbs/ft³. The stacked cement block walls have a somewhat lower density than poured structural concrete, which normally runs about 150 lbs/ft³. The density reduction means that the stacked walls must be somewhat thicker than poured concrete walls to achieve the same attenuation of radiation. The stacked walls do, however, have a very large advantage in terms of easy construction and easy removal for changes in structure.

10.6 CONTAINMENT FEATURES OF THE SHIELDS

10.6.1 CONTAINMENT OF PRIMARY PIPE AND VESSEL FAILURE ACCIDENTS. The configuration of the biological shield, and of the equipment cells beneath the reactor, is such as to insure that the reactor core will remain immersed in water in all accidents involving the structural failure of the reactor vessel or the primary piping. The reactor vessel itself is constructed with no glands or seals below the core, to prevent an accidental draining of the vessel from the failure of such components. Outside the vessel, the thermal shield and the pit in the biological shield form a second line of containment for the primary heavy water. The several pipe tunnels which pass downward from the thermal shield and the reactor vessel to the process area in the basement are all provided with seals at elevation 107 ft to preserve the leak tightness of the reactor pit. The horizontal beam tubes, which penetrate from the thermal shield cavity to the experimental area outside, are closed by heavy plugs with aluminum seal caps.

The net effect of these features of the reactor design is that no failure of the reactor vessel or of the connecting piping should lead to the active core becoming uncovered. The volume of the lower shield pit is less than the volume of heavy water contained within the reactor vessel at normal operating level. Thus, the liquid level in the reactor pit, in the event of a failure of the vessel wall, would be well above the top of the active core. In this configuration the natural circulation cooling discussed in Section 4.7.13 will take place.

In the process equipment cells beneath the reactor itself, provisions have been made to confine and direct any leakage from the primary piping into the FA-101 sump. The capacity of this sump is substantially greater than the volume of the entire primary inventory. Should a major pipe break occur anywhere in the main heavy water circulating loop, the leakage would simply collect in the process cells and drain to the FA-101 sump. The liquid level in the reactor vessel would fall to the invert of the outlet nozzle, at elevation 124 ft-1-1/2 in. To guard against the possibility of siphoning fluid from the reactor vessel, a siphon break connection is provided between the outlet and inlet nozzle regions of the reactor vessel.

The atmosphere control in the exhaust system in the building is such as to maintain a slight negative pressure in the reactor pit and in the process cells on the equipment level. This pressure difference insures that in the event of a major heavy water leak the tritium-bearing vapor will be exhausted from the reactor stack rather than diffusing throughout the building. The effects of the tritium release on the environment in such an accident are discussed in Section 14.3.4.

10.6.2 CONTAINMENT OF REACTOR EXCURSIONS. In the event of a reactor excursion, the protection of persons outside the HFBR building depends upon two lines of containment. The first of these is the shield system whose function is to contain all of the solid fragments which might be formed, and thus to protect the shell of the reactor building from damage by internal missiles. The second line of containment is the building shell itself, which serves as a gas-retaining barrier to prevent a rapid ground-level release of fission products to the environment.

The occurrence of an accident sufficiently violent to cause major disruption of the reactor system is very unlikely. The really explosive reactor accidents require very short period power bursts, so that a substantial portion of the fuel is driven to temperatures above melting before the fuel structures begin to break up. The abrupt, many-fold increase in surface area which then occurs as the molten fuel degenerates into small particles is accompanied by a large and equally abrupt energy transfer to the coolant fluid. The resulting steam formation rate may be large enough to constitute an explosion. If the chemical metal-water reactions are to enter the accident to any appreciable degree, the initiating power burst must be of even shorter period, because the threshold temperature will be higher than the melting point (1800°C for aluminum, vs. the melting point at 650°C). The destructive excursions in BORAX-1, SL-1, and SPERT-1, for instance, had periods ranging from 2.6 to 4 milliseconds and energy bursts of 31 to 135 MW-sec.

Asymptotic periods of a few milliseconds are generated in light water reactors by reactivities of a few percent. In heavy water reactors, however, much greater reactivities are needed to produce comparably short periods because of the long neutron lifetime. Even prompt jump rates of rise are relatively small in heavy water reactors for this reason. In particular, a 4 millisecond asymptotic period in the HFBR would require a reactivity insertion of almost 17%. At large reactivity insertions, the period of the prompt jump approaches the asymptotic period, and for a reactivity as large as 17% the two are nearly the same. Thus, to produce the very short periods required for explosive results, on either asymptotic or prompt jump power increases, reactivities well above the few percent supplied to the destructive light water tests would be needed. Such reactivities are not available anywhere in the HFBR system, nor are there any mechanisms available to insert large reactivities in a short time.

The type of maximum uncontrolled excursion that could be imagined in the HFBR would be characterized by a power peak of several hundred megawatts with an initial period of perhaps 50 milliseconds. Such an excursion would melt a portion of the core, and might lead to failure of the reactor vessel by overheating of the beam tubes, but it would not generate an explosive energy release.

In spite of our confidence that a violent accident will not occur in the HFBR, we have considered it prudent, in view of the essential role of the shield structure in protecting the building shell, to design into the shield a number of features to contain an explosive type of excursion. These are listed in the paragraphs below.

The outer surface of the biological shield, including the pile top in the operations area, is covered with 1 in. thick steel plate. This plate structure is assembled with full penetration welds, and is tied to the various internal steel structures in the concrete. The presence of the plate layer insures that no concrete spalls can be projected from the surface of the biological shield.

All of the experimental facility ports, horizontal and vertical alike,

have heavy steel liners cast into the heavy concrete and welded securely to the inner and outer steel plate structures of the shield. These liners terminate in 300 psi class flanges. The plugs which fill the various experimental facilities are bolted to these fixed flanges. Further, the experimental shielding requirements generally insure that there will be a great mass of heavy concrete and steel shielding outside each of the experimental holes, providing a further containment of experimental hole missiles.

The top plug structure of the biological shield is bolted together, and to the stationary steel plate structure of the shield, to form a pressure-containing member. The internal design pressure is 75 psig, with the additional provision that the bolting provides a total strain energy capacity of more than 100 million in.-lbs. The 22-5/8 in. diameter central aperture in the top plug is covered by heavy lead and steel shutters which are part of the fuel handling apparatus. These shutters are, in turn, bolted down to the top plug structure.

At the reactor pit seal floor, at elevation 130 ft-8 in., the floor structure is designed to withstand a pressure on the underside of 120 psig, again with very heavy bolting to the permanent steel structure. The intermediate pit region between the seal floor and the thermal shield is vented by two 12 in. diameter pipes which lead down into the "A" and "B" cells in the equipment area. These pressure vent pipes have 6 psi rupture disc seals at the lower ends.

The thermal shield cover, which supports the reactor vessel, is bolted to the lower thermal shield vessel. The strength of the cover member and the bolting is such as to withstand an internal pressure in the thermal shield cavity of 400 psig. The thermal shield vessel and cover form an extremely strong and massive internal shield around the reactor vessel. The thermal shield is isolated from the biological shield by a 1 in. wide gas space all around, so that it is free to expand under the loading of an accident. Venting from the thermal shield cavity is along the neck of the reactor vessel, where a 1 in. radial clearance exists between the thermal shield cover and the vessel.

The reactor vessel itself is held down by brackets bolted to the thermal shield cover, by the several coolant pipes which connect to the vessel and pass down into the process cells in the basement, and by the cavity seal floor above the inlet and outlet nozzles. This collection of retaining members will prevent the reactor vessel from being projected upward in the event of a violent excursion.

The provision of these containment features, together with unlikelihood of an explosive excursion in a reactor of the HFBR type, lead us to conclude that the building shell will remain undamaged in any accident which may occur in the reactor.

References

- (10.1) Epel, Kleimola, Notari, Rickert, and Morganthaler, Engineering Analysis of the HFBR Thermal Shield, CEND-159, Part III. (Sept., 1962).
- (10.2) Correia, Epel, Kleimola, Morganthaler, Notari, Rickert, Rohlin, and Zwickler, HFBR, Engineering Analysis of Biological Shield and Appurtenances, CEND-159, Part IV (Sept., 1962).

References (cont'd)

- (10.3) Pellini, Steele, and Hawthorne, Analysis of Engineering and Basic Research Aspects of Neutron Embrittlement of Steels, NRL Report 5780 (April 17, 1962), and Steele and Hawthorne, Neutron Embrittlement of Reactor Pressure Vessel Steels, NRL Report 5984 (October 22, 1963).
- (10.4) J. O'Donnell, "Thermal Stress Analysis, HFBR Biological Shield," BNL Memorandum (October 4, 1961).

Figure 10.1. Elevation of biological shield, thermal shield, and reactor vessel.

Figure 10.2. Elevation of thermal shield assembly.

Figure 10.3. Lower thermal shield elevation.

BILL OF MATERIALS				BILL OF MATERIALS			
ITEM NO.	NAME	NO. REQ.	MATERIAL	ITEM NO.	NAME	NO. REQ.	MATERIAL
1	Lower Thermal Shield Ass'y.	1		15	Wedge Guide	8	
2	Upper Thermal Shield Ass'y. (V-12)	1		16	Top Strip	4	ASTM-A-201-57T Grade A
3	" " " (V-10 & 11)	1		17	Side Strap	4	" "
4	Support Ring Ass'y.	1		18	Socket Head Shoulder Screw	10	ASTM-A-7-58T 1" Nom. Shoulder Dia.
5	Socket Head Cap Screw	18	ASTM-A-7-58T 3/4"-10 UNC-2Ax1 3/4" Lg.	19	Spring Lock Washer, Medium	10	ASTM-A-7-58T
6	Shim		ASTM-A-306-60T Grade 60	20	Reg. Hex. Nut	10	ASTM-A-7-58T 3/4" 10-UNC-2B
7	Reg. Hex. Bolt	28	ASTM-A-7-58T 1"-8-UNC-2Ax2 1/4" Lg.	21	Shim	6	
8	Wedge Ass'y.	12		22	Regular Hex. Bolt	6	ASTM-A-193-60T Grade B-8 3/4" 10-UNC 2Ax3-1/2" Lg.
9	" "	4		23	Washer	6	
10	Stud Bolt	16		24	Bushing	6	
11	Spherical Washer	16		25	Washer (Light)	As Req.	ASTM-A-240-58T Type 304
12	Sleeve	16		30	Square Head Pipe Plug	18	ASTM-A-105 GR. II
13	Reg. Hex. Nut	16	ASTM-A-7-58T 1-3/8" 6-UNC - 2B	31	Support	3	ASTM-A-7-58T
14	Pin	32	ASTM-A-306-60T Grade 60 - 1" Lg.				

Figure 10.4. Upper thermal shield assembly.

Figure 10.5. Heating rates in upper and lower thermal shields.

TEMPERATURE DISTRIBUTION FOR NORMAL COOLANT IN COILS

(40 MW)

TEMPERATURE DISTRIBUTION FOR LOSS OF COOLANT IN COILS

(40 MW)

Figure 10.6. Temperature distribution in the lower thermal shield for normal operation and in the event of loss of coolant in the cooling coils.

A — 3/4 INCH
B-1 — 1/2 INCH
B-2 — 1/2 INCH

C — 1 INCH
D — 4 1/2 INCHES
E — 1 INCH

BOTTOM THERMAL SHIELD

UPPER THERMAL SHIELD METAL TEMPERATURES (40 MW)

Figure 10.7. Metal temperatures in the bottom heads of the lower thermal shield, and in the plate structure of the upper thermal shield.

LOWER THERMAL SHIELD STRESSES

STRESS POINT	LOADING CONDITION		STRESS Psi
①	CYCLIC THERMAL		42,477
	STEADY STATE THERMAL		17,877
	PRESSURE		15,538
	PRESS.+ CYCLIC THERM.		15,120
	PRESS.+ STEADY THERM.		9,507
②	CYCLIC THERMAL		19,515
	STEADY STATE THERM.		9,335
	PRESSURE		1,746
	PRESS.+ CYCLIC THERM.		19,740
	PRESS.+ STEADY THERM.		10,154
③	CYCLIC THERMAL		31,052
	STEADY STATE THERM.		7,428
④	CYCLIC THERMAL		43,744
	STEADY STATE THERM.		10,280
⑤ ⑥	CYCLIC THERMAL	H-1	$\sigma_o=1,160$
			$\sigma_r=4,340$
H-4		$\sigma_o=1,395$	
		$\sigma_r=3,358$	
⑦ ⑧		H-9	$\sigma_o=2,617$
			$\sigma_r=1,753$

Figure 10.8. Stresses in the lower thermal shield for various loading conditions.

UPPER THERMAL SHIELD STRESSES

STRESS POINT	LOADING CONDITION	STRESS TYPE	STRESS (PSI)	LOADING CONDITION	STRESS POINT	STRESS TYPE	STRESS (PSI)
①	BURSTING PRESSURE	σ_o	-2,341	BURSTING PRESSURE + DEAD LOAD + PRESSURE VESSEL	⑥	σ_ϕ	31,844
	DEAD LOAD + PRESS. V.	σ_o	-1,053		⑦	σ_r	21,117
②	BURSTING PRESSURE	σ_o	-2,341		⑧	τ	8,750
		σ_r	$\pm 15,888$			σ_ϕ	19,353
	DEAD LOAD + PRESS. V.	σ_o	-1,053		⑨	σ_o	13,885
		σ_r	$\pm 7,147$			τ	11,824
③ ④	BURSTING PRESSURE	σ_o	+2,571		⑩	σ_ϕ	4,355
	DEAD LOAD + PRESS. V.	σ_o	+1,157				
⑤	BURSTING PRESSURE	σ_ϕ	50,517				
	DEAD LOAD + PRESS. V.	σ_ϕ	22,722				

Figure 10.9. Stresses in the upper thermal shield for various loading conditions.

BILL OF MATERIALS				BILL OF MATERIALS			
ITEM NO.	NAME	NO. REQ.	MATERIAL	ITEM NO.	NAME	NO. REQ.	MATERIAL
1	Biological Shield Face Plates Top	1		27	Upper Detector Tube Ass'y. (I-8)	1	
2		1		28	" " " " (I-9)	1	
3		1		29	" " " " (I-10)	1	
4		1		30	Vertical Irradiation Tube Chasse V-12	1	
5		1		31	" " " " Plate V-12	1	
6		1		32	" " " " Housing V-12	1	
7		1		34	Vertical Facilities Conduit Box	4	
8		1		35	Biological Shield Face Plates Upper	1	
9	Detector Tube Plate	1		36		1	
10		1		37		1	
11		1		38		1	
12		2		39		1	
13	Vertical Irradiation Tube Chasse V-10,V-11	2		40		1	
14	Vertical Detector Tube Service Chasse	1		41		1	
15		1		42		1	
16		1		51	Horizontal Service Chasse	8	
17		1		52	Access Pipe	1	ASTM-A-312-60T-T9304
18		1		53	" "	4	" " " " "
19	Detector Tube Locating Plate	5		55	Vertical Irradiation Tube Plate V-10, V-11	2	
20	Upper Detector Tube Ass'y. (I-1)	1		58	Housing Ass'y. Access Plug V-13,14,15 & 16	4	
21	" " " " (I-2)	1		304	Sleeve For 1/4" Hc Lines, to Inst. Hc-101C	1	
22	" " " " (I-3)	1					
23	" " " " (I-4)	1					
24	" " " " (I-5)	1					
25	" " " " (I-6)	1					
26	" " " " (I-7)	1

Chemical Lead

BILL OF MATERIALS			
ITEM NO.	NAME	QTY. REQ.	MATERIAL
15	Vertical Service Chase	1	
66	"	2	
67	Biological Shield Face Plates, Lower (H-7 & 5)	2	
87	Biological Shield Face Plates, Lower (H-8)	1	
88	"	2	(H-7 & 5)
89	"	1	(H-7)
90	"	1	(H-4)
91	"	1	(H-5)
92	"	1	(H-4)
93	"	1	(H-9)
94	"	1	(H-3)
95	"	1	(H-2)
96	"	1	(H-1)
97	"	1	(H-1)
98	"	1	(H-8)
99	"	1	(H-7)
100	"	1	(H-6)
101	"	1	(H-5)
102	"	1	(H-4)
103	"	1	(H-9)
104	"	1	(H-3)
105	"	1	(H-1)
106	Lower Horizontal Service Chase (H-1)	1	
107	"	1	(H-3)
108	"	1	(H-4)
109	"	1	(H-5)
110	"	1	(H-6)
111	"	1	(H-7)
112	"	1	(H-8)
113	"	1	(H-1)
123	Vertical Service Chase (H-5 & 7)	2	
124	"	1	(H-9 & 4)
125	"	1	(H-2 & 3)
152	Upper Inlet Housing	1	
153	Upper Outlet Housing	1	
158	Equalizing Line, Upper	1	
159	Vent Line, Upper	2	
168	Shut Down Cooling Line Sleeve, Upper	1	
169	H-2 Port Box Side Pipe March	1	
170	"	1	
171	"	1	South
172	45° Elbow-Long Radius-3" Sch. 40	1	
175	Horizontal Port & Shutter Housing Ass'y.	3	(H-1, 3&5)
176	"	2	(H-2 & 7)
177	"	1	(H-8)
179	Horizontal Port & Shutter Housing Ass'y.	1	(H-9)
181	"	1	(H-4)
182	"	1	(H-6)
183	Port Box Ass'y.	5	(H-1, 3, 5, 7 & 8)
184	"	1	(H-2)
185	"	3	(H-4 & 9)
186	Cooling Coil	1	
187	Cooling Coil	1	
188	Cooling Coil	1	
189	Cooling Coil	1	
199	P177 Upper Sleeve	1	
210	Shutter Housing Bottom Plate (H-1)	1	
211	"	1	(H-2)
212	"	3	(H-3, 5 & 7)
213	"	2	(H-4 & 6)
214	"	1	(H-8)
215	"	1	(H-9)
216	Shutter Housing Side Plate	8	
217	"	10	

Figure 10.12. Horizontal section of biological shield at elevation 118'-0".

BILL OF MATERIALS				BILL OF MATERIALS			
ITEM NO.	NAME	NO. REQ.	MATERIAL	ITEM NO.	NAME	NO. REQ.	MATERIAL
1	Plain Type Segment	1		14	Manhole Type Segment	1	
2	Dual Periscope Type Segment	1		15	Periscope Plug	4	
3	Main Top Cover Ring	1		16	Reg. Semi-Fin. Hex. Bolt	18	ASTM-A-7-58T 3/4" 10 UNC-2Aa2-1/4 Lg.
4	Special Shield Main Top Cover	1		17	Top Cover Plate	4	
5	Reg. Semi-Fin. Hex. Nut	64	ASTM-A-7-58T 1-1/2" 6 UNC-2B Std.	18	Reg. Semi-Fin. Hex. Bolt	16	ASTM-A-7-58T 1" 8 UNC-2Bx1-1/2" Lg.
6	Spring L Washer (Light)	64	ASTM-A-7-58T	19	Periscope Plug	2	
7	Main Ring Bolt	64	AISI C-1035 Hot Rolled C.S.	20	Gasket 5-1/2" O.D.x4-1/2" I.D.x1/16 Thk.	4	John Manville Asbestos Sheet Packing
8	Top Cover Segment Bolt	184		21	Gasket 8-1/2" O.D.x7-1/4" I.D.x1/16 Thk.	2	
9	Reg. Semi-Fin. Hex. Nut	184	ASTM-A-7-58T 1-3/8" 6 UNC-2B Std.	22	Reg. Semi-Fin. Hex. Bolt	30	ASTM-A-7-58T 3/4" 10 UNC-2Aa2 1/4 Lg.
10	Spring L Washer (Light)	184	ASTM-A-7-58T	23	Top Plate	4	
11	Irradiation Tube Type Segment	4		24	Top Plate	2	
12	Periscope Type Segment	1		25	Heavy Concrete Filling		
13	Top Cover Plug	1					

Figure 10.13. Biological shield top plug assembly.

BILL OF MATERIALS			
ITEM NO.	NAME	NO. REQ.	MATERIAL
1	Strut	8	ASTM-A-7-58T
2	Outer Ring	1	"
3	Inner Segment	8	"
4	Outer Center Segment	8	"
5	Inner Ring	1	"
6	Top Center Plate	1	"
7	Top Plate	1	"
8	I-Beam Cover Plates	16	"
9	Plate Blocks	136	"
10	Bottom Outer Plate	1	"
11	Stud Sleeve	64	ASTM-A-53-58aT Grade B
12	Stiffeners	128	ASTM-A-7-58T
13	Inner Segment Web	8	"
14	Bottom Center Plate	1	"
15	Plate Blocks	24	"

Figure 10.14. Biological shield top plug support frame.

Figure 10.15. Radial thermal flux plots in the reactor vessel neck.

Figure 10.16. Axial thermal flux plots in the reactor vessel neck.

Figure 10.17. Sodium-24 concentration in the primary D_2O and gamma dose rate at the reactor vessel top flange from sodium-24.

SECTION 11. RADIOACTIVE WASTES AND EFFLUENTS

11.1 GASEOUS WASTES

The gaseous radioactive materials listed in the sections below are formed in the operation of the HFBR and are vented from the system through the reactor stack. For the most part, these gases are present in normal operation in small amounts and offer no particular problem. The tritium content of the primary coolant is quite large after extended operation, however, and special attention is given to this component of the exhaust gas.

11.1.1 TRITIUM-BEARING GASEOUS EXHAUSTS. Tritium is the hydrogen isotope of mass 3. It is formed by neutron capture in deuterium in the core and reflector of the reactor. The tritium is held almost entirely in water molecules of the type TDO, so that the gaseous waste problem is concerned with heavy water vapor containing traces of the tritium-bearing molecular species. The tritium half-life is 12.26 years. The emission from tritium is a beta particle with an average energy of 5.7 Kev. The concentration of tritium in the primary system as a function of operation time is given in Figure 5.7, Section 5.2.4. For the calculations of this section a tritium concentration of $1.7 \times 10^4 \mu\text{c}/\text{cm}^3$ heavy water is assumed. This concentration is the 25-year level which would be reached if no heavy water were lost and replaced in the primary system.

11.1.1.1 Reactor Cover Gas Purge During Operation. The helium cover gas in the reactor vessel is purged at a slow rate to control the impurities (see Section 7.4). The expected purge rate is 0.01 cfm at system temperature and pressure (120°F , 200 psig). Tritium-bearing heavy water vapor will accompany the purge gas through the exhaust system and be discharged from the stack. At 120°F the partial pressure of heavy water vapor is 1.55 psia. The heavy water loss rate at 0.01 cfm is $30.4 \text{ cm}^3/\text{day}$, or 10.0 liters/year. The tritium activity exhaust rate is $5.97 \mu\text{c}/\text{sec}$ (11.1).

The ground level tritium concentrations in air are given in Table 11.1-1 for the four meteorological conditions discussed in Section 2.3.2. The concentrations given are the maximum values for each of the diffusion cases. The tritium levels are about 10^{-4} of the maximum permissible concentration given in AECM-0502, Annex 1 (11.2), which is $2 \times 10^{-7} \mu\text{c}/\text{cm}^3$ air.

11.1.1.2 Surge Volume Depressurization. At the end of each operating cycle the helium cover gas is vented to the stack to reduce the primary system pressure. Heavy water vapor in the gas space is thus exhausted from the stack. The helium system total volume is 27 ft^3 at operating conditions. The venting takes several minutes. More rapid depressurization occurs when the flow reversal system is tripped (see Section 14.3.5). The most rapid depressurization expected of the system takes 4 seconds. We assume the latter value, and a temperature of 120°F . The heavy water content of the cover gas is about 55 cm^3 . The tritium content is 0.97 curies and the activity release rate is $0.24 \text{ c}/\text{sec}$ (11.1).

The ground level tritium concentrations in air, the integrated cloud exposures, and the integrated doses are given in Table 11.1-2 for the four meteorological conditions discussed in Section 2.3.2. The quantities given are the maximum values for each condition. The conversion from tritium cloud exposure (in $\text{c}\cdot\text{sec}/\text{m}^3$) to dose (in rem) is made by the method of Pinson and Langham, using a breathing rate of 20 liters/min and a quality factor of

1.7 rem/rad for the 5.7 Kev beta particle (11.4).

Table 11.1-1 Maximum Ground Level Tritium Concentrations
from Normal Operating Cover Gas Purge

Atmospheric Condition	Maximum Ground Concentration ($\mu\text{c}/\text{cm}^3$ air)	Distance of Maximum Concentration (meters)
I. Weak neutral, inversion lid at stack top	6.0×10^{-11}	2600
II. Weak neutral	3.0×10^{-11}	2600
III. Strong inversion, typical night	2.2×10^{-12}	40,000
IV. Normal lapse, typical day	1.7×10^{-11}	600

Table 11.1-2 Maximum Ground Level Tritium Doses
from Cover Gas Depressurization

Atmospheric Condition	Max. Ground Concentration ($\mu\text{c}/\text{cm}^3$ air)	Max. Cloud Exposure (c-sec/ m^3)	Maximum Dose (rem)
I. Weak neutral, inversion lid at stack top	2.4×10^{-6}	9.7×10^{-6}	1.3×10^{-6}
II. Weak neutral	1.2×10^{-6}	4.8×10^{-6}	6.4×10^{-7}
III. Strong inversion, typical night	8.7×10^{-8}	3.5×10^{-7}	4.6×10^{-8}
IV. Normal lapse, typical day	6.8×10^{-7}	2.7×10^{-6}	3.6×10^{-7}

The distances for the maximum doses given in Table 11.1-2 are the same as those in Table 11.1-1 for the same meteorological cases.

11.1.1.3 Evaporation During Fuel Changes. At each scheduled shutdown the vessel is opened and half the fuel charge is removed to the canal. The estimated heavy water loss during fuel changing due to evaporation from the liquid free surface and from fuel elements raised out of the vessel is one gallon. The vessel is open for about 8 hours in the fuel changing operation, but the heavy water loss is concentrated in a shorter time, and a release interval of 3 hours is used for these calculations. The total tritium activity release is 64.3 curies at a rate of 6.0×10^{-3} c/sec. Table 11.1-3 gives the maximum ground level concentrations, cloud exposures, and total doses for the four

meteorological conditions discussed in Section 2.3.2. The distances at which the maximum doses occur are the same as in Table 11.1-1 for the same meteorological conditions.

Table 11.1-3 Maximum Ground Level Tritium Doses from Heavy Water Loss in Fuel Changing

Atmospheric Condition	Max. Ground Concentration ($\mu\text{c}/\text{cm}^3$ air)	Max. Cloud Exposure ₃ (c-sec/ m^2)	Maximum Dose (rem)
I. Weak neutral, inversion lid at stack top	6.0×10^{-8}	6.4×10^{-4}	8.5×10^{-5}
II. Weak neutral	3.0×10^{-8}	3.2×10^{-4}	4.3×10^{-5}
III. Strong inversion, typical night	2.1×10^{-9}	2.3×10^{-5}	3.1×10^{-6}
IV. Normal lapse, typical day	1.7×10^{-8}	1.8×10^{-4}	2.4×10^{-5}

11.1.1.4 Valve and Pump Seal Leakage. An estimate is made in Section 5.2.3 of the amount of heavy water leakage from valve packings and pump and control rod shaft seals which is vaporized and exhausted from the stack. The annual loss from these sources may be as much as 40 to 80 gallons. Table 11.1-4 shows the maximum ground level concentrations of tritium from this source, on the assumption of a constant 80 gallon annual loss. The concentrations are about 10^{-2} of the AECM-0502 maximum permissible concentration (11.2), which provides an ample margin for short-term, higher loss rate occurrences.

Table 11.1-4 Maximum Ground Level Tritium Concentrations from Valve and Pump Seal Leakage

Atmospheric Condition	Maximum Ground Concentration ($\mu\text{c}/\text{cm}^3$ air)	Distance of Maximum Concentration (meters)
I. Weak neutral, inversion lid at stack top	1.6×10^{-9}	2600
II. Weak neutral	8.2×10^{-10}	2600
III. Strong inversion, typical night	5.9×10^{-11}	40,000
IV. Normal lapse, typical day	4.6×10^{-10}	600

11.1.2 VOLATILE FISSION PRODUCT EXHAUST. In the normal operation of the reactor trace amounts of fission products will be present in the primary coolant. The volatile fission products will collect in the reactor surge volume and will be purged with the cover gas. The point of discharge from the reactor complex is the stack. The source of the fission products in normal operation, i.e., in the absence of fuel element failures, is the surface contamination of the elements by U-235. This contamination, which occurs in the fabricator's plant, is small but unavoidable. Measurements indicate that the average U-235 surface contamination to be expected is about $1/4 \mu\text{gm}$ per square foot of fuel plate surface, and that the contamination is seldom more than $1 \mu\text{gm}/\text{ft}^2$ (11.5).

For the purpose of determining the volatile fission product exhaust from surface contamination, $1 \mu\text{gm}/\text{ft}^2$ of fuel plate is assumed. The total fuel plate area in the core is 432 ft^2 , giving a total U-235 surface inventory of $432 \mu\text{gm}$. The fission product production from the surface U-235 is assumed to be the same, per unit mass of U-235, as for the core at maximum burnup (11.6). The total accumulation of the volatile fission products for the two-cycle life of the elements is shown in the first column of Table 11.1-5, and the surge volume inventory during operation is shown in the second column. The corresponding concentrations of the various isotopes are shown in the third column of the table. There are several possible release conditions for the fission products which are discussed in the sections which follow.

Table 11.1-5 Volatile Fission Products in the Reactor
Surge Volume in Normal Operation

Isotope	Total Accumulation in 2 Cycles (curies)	Surge Volume Normal Inventory (μc)	Surge Volume Concentration at 120°F , 200 psig ($\mu\text{c}/\text{cm}^3$)
I-131	.0517	2420	.0032
I-132	.0867	4070	.0053
I-133	.125	5860	.0077
I-134	.164	7690	.0101
I-135	.112	5250	.0069
Kr-83 ^m	.0094	440	.00058
Kr-85 ^m	.0318	1490	.0020
Kr-87	.0531	2490	.0033
Kr-88	.0707	3320	.0043
Xe-131 ^m	.00044	20	.00003
Xe-133 ^m	.00303	142	.00019
Xe-133	.132	6190	.0081
Xe-135 ^m	.0354	1660	.0022
Xe-135	.0189	886	.0012

11.1.2.1 Reactor Cover Gas Purge During Operation. The volatile fission product inventory from surface contamination of the elements, listed in Table 11.1-5, is vented with the cover gas purge at a uniform rate during normal operation. The exhaust gas passes through the filters FD-105 and FD-105A, which are a 5-micron stainless steel particle filter and an activated charcoal bed, respectively (see Section 7.4.4). The gas then passes to the off-gas manifold, through the building exhaust air filters, and is discharged from the stack. An iodine removal efficiency for the charcoal filter of 95% is assumed, although tests of elemental iodine removal by charcoal beds have shown typical efficiencies of 99.9% or better. No other filter effects are included. The normal cover gas purge results in the fission product stack release rates given in Table 11.1-6. The ground level concentrations of the various isotopes are also given in the table, for the maximum concentration point in meteorological Case I (see Section 2.3.2). The Case I maximum ground concentration distance is 2600 meters. Cases II, III, and IV are not listed but may be inferred from Table 11.1-6 by multiplying by factors of 0.5, 0.036, and 0.28, respectively. The last column of Table 11.1-6 lists the AECM-0502 maximum permissible concentrations for the various isotopes (11.2).

Table 11.1-6 Volatile Fission Product Maximum Ground Concentrations from Normal Cover Gas Purge

Isotope	Stack Release Rate ($\mu\text{c}/\text{sec}$)	Case I Max. Ground Concentrations ($\mu\text{c}/\text{cm}^3$ air)	AECM-0502 Max. Permissible Concentration ($\mu\text{c}/\text{cm}^3$ air)
I-131	.00075	7.5×10^{-15}	3×10^{-10}
I-132	.0013	1.3×10^{-15}	8×10^{-9}
I-133	.0018	1.8×10^{-14}	1×10^{-9}
I-134	.0024	2.4×10^{-14}	2×10^{-8}
I-135	.0016	1.6×10^{-14}	4×10^{-9}
Kr-83 ^m	.0027	2.7×10^{-14}	
Kr-85 ^m	.0092	9.2×10^{-14}	1×10^{-7}
Kr-87	.015	1.5×10^{-13}	2×10^{-8}
Kr-88	.021	2.1×10^{-13}	
Xe-131 ^m	.00013	1.3×10^{-15}	4×10^{-7}
Xe-133 ^m	.00088	8.8×10^{-15}	
Xe-133	.038	3.8×10^{-13}	3×10^{-7}
Xe-135 ^m	.010	1.0×10^{-13}	
Xe-135	.0055	5.5×10^{-14}	1×10^{-7}

11.1.2.2 Surge Volume Depressurization. The venting of the surge volume pressurizing gas at the end of each operating cycle releases the fission products in the gas. The venting path is through the charcoal filter, and an iodine filtration efficiency of 95% is assumed. Emergency depressurization, on the other hand, vents the surge volume directly to the off-gas manifold,

bypassing the charcoal filter. In this case the iodine removal efficiency of building exhaust air filter is assumed to be 70%. (No credit for the building exhaust filter is taken in cases where the gas passes through the charcoal filter first, because any iodine passing the charcoal filter is unlikely to be in the elemental form, and the building filter efficiency for non-elemental iodine is unknown.) The two depressurization cases are treated by calculating the doses from the larger release (70% filter efficiency for iodine). The normal depressurization case iodine doses will be a factor of 6 less for the assumed charcoal filter efficiency 95% vs. 70%) and as much as a factor of 300 less for the expected efficiency (99.9% vs. 70%). The rare gas release is the same in all cases.

The quantities of the various isotopes released may be taken from Table 11.1-5, the "Surge Volume Normal Inventory" column, with a factor of 0.30 on the iodine numbers to account for the assumed filtration. The thyroid and whole body gamma doses which result are given in Table 11.1-7 for the four meteorological conditions of Section 2.3.2. The conversions from cloud exposure to dose are given in Table 14.6-1 in Section 14.6.2.3.

Table 11.1-7 Maximum Ground Level Thyroid and Whole Body Gamma Doses from Surge Volume Emergency Depressurization

Atmospheric Condition	Max. Thyroid Dose (rad)	Max. Gamma Dose (rad)	Dist. of Max. Doses (meters)
I. Weak neutral, inversion lid at stack top	7.3×10^{-6}	5.0×10^{-8}	2600
II. Weak neutral	3.7×10^{-7}	2.5×10^{-8}	2600
III. Strong inversion, typical night	2.7×10^{-7}	1.8×10^{-9}	40,000
IV. Normal lapse, typical day	2.1×10^{-6}	1.4×10^{-8}	600

11.1.3 OTHER SURGE VOLUME RADIOACTIVE GASES. The table of radioactive contaminants expected in the primary coolant, Table 5.2-2 of Section 5.2.4, lists several gases. The isotopes of interest are N-16, N-17, O-19, and F-17. Traces of these gases will appear in the surge volume, and will be released from the stack with the normal cover gas purge and with the depressurizing of the surge volume. The isotopes are present in amounts which are small compared to the tritium burden in the coolant, e.g., $100 \mu\text{c}/\text{cm}^3$ for N-16 compared to $17,000 \mu\text{c}/\text{cm}^3$ for tritium. The concentrations to be expected from the stack release will be correspondingly smaller than the tritium concentrations given in Section 11.1.1. Further, the half-lives of the oxygen and nitrogen isotopes are short, 4.2 to 29 seconds, and the decay in the several minutes which are required for the gases to pass through the building exhaust system and stack and diffuse back to the ground will reduce the concentrations at ground level substantially. The slightly longer-lived F-17, with a 67 second half-life, should be filtered quite efficiently in the charcoal bed.

11.1.4 SHIELD CAVITY GAS PURGE. The lower pit of the reactor shield has

a CO₂ atmosphere which is at about 170°F and 1 atmosphere pressure during operation. A purge rate of 0.5 cfm is maintained through the cavity, with the exhaust gas discharged from the reactor stack (see Section 7.7). Of the total cavity volume of 1200 ft³, a net volume of 133 ft³ of gas exists within the thermal shield and in the beam tubes where the thermal neutron flux is high enough to produce measurable quantities of carbon-14 and argon-41. The average thermal flux in the active volume is about 1×10^{13} n/cm²-sec.

11.1.4.1 Carbon-14 Activation. The long-lived (5730 year half-life) carbon isotope C-14 is produced in the active volume of the shield cavity by the thermal neutron interactions C-13 (n,γ) C-14, with cross section 0.9 millibarns, and N-14 (n,p) C-14, with cross section 1.75 barns. The nitrogen appears in the pit gas as an impurity in the basic CO₂ atmosphere. The carbon dioxide contains about 0.5% air by volume, giving a nitrogen concentration of 1.7×10^{17} atoms/cm³. The larger cross section for the nitrogen activation causes this process to dominate the carbon-14 production. The total macroscopic activation cross section is 2.9×10^{-7} cm⁻¹.

The residence time of the gas in the active volume of the pit is 1.6×10^4 sec, and the activation is directly proportional to this time since the half-life is very long. The carbon-14 activity in the cavity gas is 4.8×10^{-6} μc/cm³, and at a purge flow of 0.5 cfm the release rate is 1.1×10^{-3} μc/sec. The maximum ground level concentration from the stack discharge is 1.1×10^{-14} μc/cm³ air, for meteorological Case I (see Section 2.3.2). The AECM-0502 maximum permissible concentration in air is 1×10^{-7} μc/cm³ (11.2).

When beam tube seal plugs are to be replaced at a shutdown, it may be desirable to ventilate the cavity for several hours at an increased flow rate. The ventilation flow might be as high as 20 cfm, which would increase the release rate and ground level concentration by a factor of 40 for a limited time.

11.1.4.2 Argon-41 Activation. The radioisotope A-41 is formed from A-40 by thermal neutron capture with an activation cross section of 0.53 barns. The naturally occurring argon isotope A-40 is present in air to the extent of 0.94%. Air is a 0.5% impurity in the cavity CO₂ atmosphere. At normal operating temperature and pressure in the shield cavity the A-40 concentration is 0.99×10^{15} atoms/cm³ and the macroscopic activation cross section is 5.2×10^{-10} cm⁻¹. The half-life of A-41 is 109 minutes.

The residence time of the cavity gas in the active region (4.4 hours) is long enough compared to the A-41 half-life to assume equilibrium. The active region equilibrium activity is 0.14 μc/cm³ gas. At a purge flow of 0.5 cfm the argon-41 activity release rate from the active volume is 33 μc/sec. The exhaust to the stack is from the upper part of the cavity space, with a net volume of about 1070 ft³. The average residence time of the purge gas in the inactive (i.e., low neutron flux) cavity volume is 35-1/2 hours. The A-41 decay factor in this time is 1.23×10^{-6} , and the exhaust gas to the stack has an activity of only 1.73×10^{-7} μc/cm³. The corresponding activity release rate is 4.1×10^{-5} μc/sec. The maximum ground level concentration which results from the stack discharge is 4.1×10^{-16} μc/cm³ air, for meteorological Case I (see Section 2.3.2), as compared to the AECM-0502 maximum permissible concentration of 4×10^{-8} μc/cm³ air (11.2).

If the cavity purge rate is increased for ventilation purposes at a shutdown, the argon-41 discharge is increased not only by the greater volumetric flow but also by the reduced residence time of the gas in the inactive volume of the cavity. The maximum effect would occur if the ventilation were done

immediately after shutdown, before the argon decay had progressed very far. At 40 times the normal flow rate the complete cavity volume is replaced in an hour. The maximum increase in the argon-41 release rate is 2.3×10^7 , with a factor of 40 from the volumetric flow and a factor of $.57 \times 10^6$ from the reduced decay time. Even with this large increase, the maximum ground level concentration is only $.95 \times 10^{-8}$ $\mu\text{C}/\text{cm}^3$ air, or about 25% of the maximum permissible concentration.

11.2 LIQUID WASTES

11.2.1 TRITIUM-CONTAMINATED LIQUID WASTE. A portion of the radioactive liquid waste accumulated in the operation of the plant will contain heavy water and will be contaminated with tritium. The sources of such wastes are heavy water leakage from pump shaft seals and valve packings, heavy water spills, ion bed deuteration and dedeuteration fluids, and experimental sample capsule fill water. The tritium-contaminated wastes are treated according to the heavy water content. Where a waste fluid has enough heavy water content to make reprocessing economical, the waste will be filtered and demineralized to reduce the radioactive and chemical impurities (except tritium) to an acceptable level. If the light water content of the waste is low enough, it may be reused in HFBR operations. If the light water content is too high for HFBR use, the purified waste is returned to the Savannah River AEC plant for upgrading.

The residual tritium-contaminated liquid waste which cannot be recovered for further use will be used as mix water in concrete. The concrete is cast into the form of inner linings of the disposal vaults which are used at Brookhaven for solid waste disposal. The quantity of tritium-contaminated liquid waste to be treated in this manner should be small since the high cost of heavy water makes recovery an economical process even if several cycles of filtering and demineralization are necessary for decontamination. The ion bed resins themselves are likely to be the only tritium-contaminated wastes to be cast in concrete.

11.2.2 OTHER LIQUID WASTES. Liquid wastes which are not tritium-contaminated are collected throughout the reactor building in the D and F waste systems (see Section 3.5.2). The D wastes are those which are almost certainly contaminated by radioactive material. The D wastes are collected in the GA-304 pump sump in the basement of the building and are pumped to the Hot Laboratory waste storage tanks. After sampling and whatever chemical treatment is necessary, the wastes are pumped to the BNL Waste Treatment Area. There they are distilled to reduce the liquid volume. The discharged concentrate from the treatment plant is solidified into the waste disposal vaults.

The F wastes are those which may possibly contain radioactive material. The F wastes are collected in the storage tank FA-305 in the basement of the HFBR building. Samples are taken from the liquid in FA-305 and are analyzed for radioactive material content. Waste water which is below tolerance in all respects is drained to the sewer. Radioactive F waste above the discharge tolerance is drained to the D system and is pumped to the Hot Laboratory for disposal.

11.3 SOLID WASTES

Radioactive solid waste material with a surface activity of less than 50 mr/hr is collected in special containers in the reactor building. The containers are periodically removed to the BNL waste storage area and emptied into concrete disposal vaults with the other solid wastes from laboratory operations. Solid materials of high activity are placed in shielded containers

and are also removed to the concrete vaults.

Arrangements have been made for permanent storage of the concrete vaults in the New York State burial ground near Buffalo, New York. The W. R. Grace Chemical Co. operates this facility in connection with the fuel element chemical processing plant now being built there.

Spent fuel element canal storage and transfer to the processing plant is discussed in Section 7.8.5.

11.4 PRIMARY-SECONDARY LEAKAGE

The primary coolant pressure in the main and shutdown heat exchangers is always higher than the secondary water pressure in the units (see Sections 5.4.7 and 9.4.3.2). The pressure difference insures that in the event of a leak in the exchangers the heavy water flows into the secondary system. This feature of the system is necessary to guard against diluting the primary coolant with light water. At the same time, the possibility of heavy water leakage into the secondary system must be recognized. The possible presence of traces of heavy water in the secondary system requires a careful examination of the tritium levels to be expected. The sections which follow deal with this problem.

11.4.1 LEAKAGE IN THE AS-BUILT COOLERS. There are four heat exchangers in the plant with heavy water on one side and secondary water on the other. Three of these are in the primary system; the main coolers EA-101A and EA-101B, and the shutdown cooler, EA-103. The fourth unit is the experimental facilities heavy water system cooler, EA-102. All of the coolers were tested for tube-side to shell-side leakage by mass spectrometer measurements with helium at 20 to 50 psig on one side and vacuum on the other. The test results showed no leaks exist in any exchanger which are larger than the sensitivity of the mass spectrometer. The spectrometer sensitivity in these tests was 3.5×10^{-10} STP $\text{cm}^3 \text{He/sec}$.

Calculations have been made to determine the maximum primary-to-secondary leakage, based on the mass spectrometer measurements. For these calculations it was assumed that each exchanger had a leak corresponding to 1×10^{-9} STP $\text{cm}^3 \text{He/sec}$, and that the configuration of the leak path was such as to give a maximum flow for incompressible fluids. The resulting leak path is an orifice of area $6.2 \times 10^{-18} \text{ft}^2$ (11.7). The heavy water leakage through such a leak path at a pressure difference of 250 psig is $3.2 \times 10^{-11} \text{cm}^3/\text{sec}$.

The equilibrium tritium content of the secondary water from the leakage from all four exchangers is a function of the HFBR operating time. Figure 5.7 of Section 5 gives the tritium buildup in the primary system. At very long operating times, 25 years, with no heavy water makeup to the primary system, the tritium content is $1.7 \times 10^4 \mu\text{c}/\text{cm}^3 \text{D}_2\text{O}$. The corresponding secondary water concentration is $4.3 \times 10^{-11} \mu\text{c}/\text{cm}^3$, compared to the maximum permissible concentration of $3 \times 10^{-3} \mu\text{c}/\text{cm}^3$ given in AECM-0502 (11.2). Thus, the "as built" heat exchanger leakage presents no contamination problem in the secondary water, even on the basis of the very conservative calculation which has been made.

In the operation of the plant only the three primary system coolers are of interest in the primary-secondary leakage problem. The experimental facilities cooler operates with a higher secondary side pressure, and any leakage in that unit would be toward the heavy water side.

11.4.2 DETECTION OF HEAVY WATER LEAKAGE. Although the heat exchangers

will start operation with essentially zero leakage, it is possible that leaks may develop in the future. The kind of leaks to be expected are corrosion cracks, pinholes in tube longitudinal welds, or small holes at wear points under baffles. Once started, such leaks may increase with time and may eventually end in tube failures if no repairs are made. The early detection of small leaks is, therefore, important in maintaining the exchangers.

Two methods of primary-secondary leakage detection are provided in the reactor instrumentation. The first is a continuous monitoring of the secondary effluent from each exchanger for radioactivity. The instruments are scintillation counters set into wells in the secondary pipes (see Section 9.6.2).

The scintillation instruments give a high activity alarm at a secondary water contamination level of $2.4 \times 10^{-5} \mu\text{C}/\text{cm}^3$ of 1 Mev gamma rays. The activity detected is mainly from the N-16 content in the D_2O . The heavy water N-16 content is 40 to $100 \mu\text{C}/\text{cm}^3$. The higher concentration of N-16 has been used for shielding studies, but the lower value is of interest here to establish the minimum leak detection sensitivity. The decay of the N-16 between the reactor core and the point of detection is also an important factor. The passage time ranges from 10 to 30 seconds, depending on the location of the leak. Using the minimum N-16 concentration in the primary coolant and the maximum decay time, the leak rate which gives an alarm from the scintillation instrument is about $2.5 \text{ cm}^3/\text{sec}$ for the main coolers and $0.1 \text{ cm}^3/\text{sec}$ for the shutdown cooler. The detection sensitivity is much better in the shutdown cooler because the secondary flow rate, hence the dilution of the N-16 activity, is much less (150 gpm vs. 4000).

During reactor shutdowns the N-16 activity disappears and the leak detection depends upon Na-24 in the heavy water. The shutdown cooler is the only operating heat exchange unit at such times, and the combination of the lower secondary flow, the lower heavy water gamma activity, and the longer half-life yields a leak rate for alarm of about $1.5 \text{ cm}^3/\text{sec}$ soon after shutdown. Later in the shutdown period the primary system sodium activity is greatly reduced by decay and removal in the purification system and the scintillation instrument becomes ineffective for leak detection.

The second method for leak detection is by periodic sampling and analysis of the secondary effluent from each exchanger. In this method the tritium concentration in the secondary water samples is measured directly. The low energy (5.7 Kev average) of the beta particle from tritium makes the analysis difficult. The water samples are filtered, reduced to a standard volume by pipetting, and diluted with a standard volume of liquid scintillator solution. The samples are then chilled and kept away from light for about 30 minutes to reduce spurious scintillator activity. Counting times of 10 minutes give a sufficient total count to detect a tritium concentration of about $1 \times 10^{-4} \mu\text{C}/\text{cm}^3$ in the original sample. The sampling and analysis is carried out at the beginning of each 8-hour shift on a routine basis. The results are available about 2 hours after the samples are taken. The sample treatment and counting systems are installed in the heavy water laboratory on the operations level of the building.

The great sensitivity of the laboratory analysis method means that very small leaks can be detected. The tritium activity leakage rate which gives a minimum detectable concentration in the secondary water is $25 \mu\text{C}/\text{sec}$ for the main exchangers and about $1 \mu\text{C}/\text{sec}$ for the shutdown cooler. At the 25-year tritium concentration of $1.7 \times 10^4 \mu\text{C}/\text{cm}^3$ in heavy water, the minimum detectable leak rate is $0.0015 \text{ cm}^3/\text{sec}$ in the main exchangers. Even at one year, with $1250 \mu\text{C}/\text{cm}^3$ tritium concentration in the primary, the minimum detectable

leak is only $0.02 \text{ cm}^3/\text{sec}$.

The corresponding minimum detectable leak rates for the shutdown cooler are a factor of about 25 less, $0.0008 \text{ cm}^3/\text{sec}$ at one year and $0.00006 \text{ cm}^3/\text{sec}$ at 25 years. These leak rates are calculated at the beginning of a leak, so that no credit is taken for the buildup of tritium in the secondary system. The secondary buildup to equilibrium takes about 12 hours, and reduces the minimum detectable leak rates for the main exchangers by a factor of about 5.

11.4.3 SECONDARY SYSTEM TRITIUM CONCENTRATION. In the event of a primary-to-secondary leak the reactor would be shut down and the secondary system blow down stopped. The return pump, GA-303, for secondary water from the shutdown cooler would be started, eliminating the water discard to the storm sewer. The offending exchanger would be closed off on both primary and secondary sides. The nature of the leaks which might be expected and the sensitivity of the sample analysis method of leak detection make it likely that the leakage could be controlled before the secondary water tritium content reached tolerance levels. There is, however, the possibility of a leak at a rate just below the detection limit for the "on-line" monitors continuing for the 8-hour interval between sample analyses. Such a leak would give a maximum tritium burden in the secondary system. The concentrations in the secondary water in this case have been calculated as a function of the operating time of the reactor (11.8). The results are given in Table 11.4-1. A main heat exchanger leak rate of $2.5 \text{ cm}^3/\text{sec}$ was assumed. In addition to the concentration in the secondary water at the end of the full 8-hour leakage period, the average concentration during the period is given, and also the concentration at full equilibrium.

11.4.4 POINTS OF RELEASE OF SECONDARY WATER, CONCENTRATIONS AND DOSES. In the event of a heavy water leak to the secondary system the various points of release of secondary water must be examined and the resulting concentrations and doses determined. For these purposes the secondary water tritium burdens given in Table 11.4-1 are assumed. The points of release to be considered are evaporation from the basin, the normal operating tower blow down to the storm sewer and to the ground, and direct leakage from the basin.

Table 11.4-1 Secondary System Tritium Concentrations from
a $2.5 \text{ cm}^3/\text{sec}$ Heavy Water Leak

HFBR Operating Time (years)	Concentration at End of 8 Hours ($\mu\text{C}/\text{cm}^3$)	Average Concentration During 8 Hours ($\mu\text{C}/\text{cm}^3$)	Concentration at Equilibrium ($\mu\text{C}/\text{cm}^3$)
0.5	.027	.018	.031
1.0	.055	.037	.062
1.5	.080	.053	.089
2.0	.11	.071	.12
3.0	.16	.11	.18
25.0	.75	.50	.84

11.4.4.1 Evaporation. The maximum tritium concentration in air around the cooling towers may be determined by assuming 100% relative humidity due to the

tower water. An air temperature of 80°F and the "25-year" tritium concentration in the basin of $0.75 \mu\text{g}/\text{cm}^3$ are assumed. At these conditions the air contains $2.68 \times 10^{-5} \text{ cm}^3 \text{ water}/\text{cm}^3 \text{ air}$, and the tritium concentration is $2.0 \times 10^{-5} \mu\text{g}/\text{cm}^3$ air. The concentration is somewhat above even the 40-hour occupational concentration of $5 \times 10^{-6} \mu\text{g}/\text{cm}^3$ (11.3), and careful monitoring of the actual concentration and of working time would be necessary for any extended work around the towers. The concentration would decrease rapidly with distance from the basin.

11.4.4.2 Secondary System Blow Down. The normal operation of the secondary system entails a total blow down of about 500 gpm to stabilize the solids content of the system (see Section 6.5). Most of the blow down (350 gpm) is piped to an unoccupied location east of the reactor and is released to the ground. The balance of the blow down is taken by dumping the effluent from the shutdown cooler into a storm sewer. Like the direct blow down, the storm sewer ends in an open basin to the east of the reactor. In the event of a heavy water leak to the secondary system both blow down paths would be closed off as soon as the leak was detected. The shutdown heat rejection would be maintained through the shutdown cooler by using domestic water on the secondary side instead of tower water. If the shutdown cooler itself were the leaking unit, some flow would be maintained through a main exchanger for cooling.

Since it is possible for a heavy water leak to go undetected for 8 hours, the tower blow down would continue during this period, with tritium concentrations in the released water as given in Table 11.4-1 for the maximum leak of $2.5 \text{ cm}^3/\text{sec}$. A minor effect of the release would be local air concentrations of tritium which might be above tolerance. However, the dump basin regions are remote from normally occupied areas on the site and the condition should not last for more than a few days. A much more serious effect of the release would be the addition of above-tolerance tritiated water to the ground water.

The blow down streams carry enough flow so that a large proportion of the released water reaches the upper level of the water table. The higher temperature of the blow down (compared to the ground water) probably causes it to remain at the surface of the water table and to spread horizontally. As the ground water moves slowly to the southeast the blow down layer is diluted by rainfall additions. The rate of travel is 0.5 to 1 ft/day (see Section 2.4), and the distance to the site boundary is about 3500 ft. The contaminated blow down water from the 8-hour period of undetected leakage would eventually move off-site and might be picked up in wells to the southeast of the laboratory.

An upper limit to the tritium doses which might result from the blow down can be set by assuming slug flow in the water table, with no dilution or decay. In this model the block of contaminated water moves past a well point which draws the full blow down concentration for the 8-hour release period. Tritium body burdens and total doses calculated on this basis are given in Table 11.4-2 (11.8). The average 8-hour tritium concentrations in the secondary water (Table 11.4-1) are used in this calculation, and the results are given as a function of the time the HFBR has operated. The body burden is based on a water rate of 2.2 liters/day, and the conversion from body burden to dose is $0.198 \text{ mrem}/\mu\text{g}$ (11.4).

11.4.4.3 Leakage from the Cooling Tower Basin. The direct leakage from the cooling tower basin is estimated to be 10 gallons/hr at most, based on tests on the new concrete work at the towers. This leakage goes into the ground under the basin. Test borings which were made as part of the foundation study for the HFBR showed the water table to be at least 90 ft below the bottom of the basin. The volume of the sand bed into which the basin leakage flows is,

therefore, very large. The leak rate is relatively small and five or six years elapse before the leakage reaches the water table. The quantity of contaminated water released in direct leakage from the basin, in the event of a heavy water leak, is small compared to the blow down release. Even if several months are required to dispose of the contaminated basin water, the direct leakage from the basin amounts to less than 10% of the 8-hour blow down.

Table 11.4-2 Tritium Body Burdens and Doses from Maximum Blow Down Release to Ground Water

HFBR Operating Time (years)	Body Burden (μc)	Total Dose (mrem)
0.5	13	2.7
1.0	27	5.4
1.5	39	7.7
2.0	52	10.
3.0	78	15.
25.	370	73.

11.4.4.4 Contaminants Other Than Tritium. The doses and concentrations due to the release of contaminated secondary water have all been given for tritium in the sections above. A heavy water leak from the primary to the secondary will carry not only tritium but other primary system contaminants, and the effects of these must be considered. The tritium doses and concentrations, however, are controlling because soon after the reactor commences operation the tritium concentration becomes much larger, compared to tolerance concentrations, than that of any other radioactive contaminant. At full tritium content, for instance, the ratio of primary water tritium concentration to the tolerance concentration is $1.7 \times 10^4 / 3 \times 10^{-3}$, or about 5.7×10^6 . In contrast, the sodium primary water concentration (Na-24, Table 5.2-2) is only 90 times tolerance, and is reduced to well below tolerance by dilution in the secondary system. Similarly, the long-lived fission products from surface U-235 contamination of the fuel elements are expected to have primary water concentrations which are within a factor of 10 of the permissible levels, and these contaminants would have below-tolerance concentrations in the secondary water.

11.4.5 DISPOSAL OF CONTAMINATED SECONDARY WATER. The disposal of the large volume of contaminated secondary water which would result from a heavy water leak is a special case which is not covered by the normal tritium-bearing liquid waste procedures discussed in Section 11.2.1. The large volume involved, 175,000 gallons, and the low heavy water content make it uneconomical to attempt to recover the heavy water. The disposal method proposed for the secondary water is to evaporate it, with the vapor being discharged from the reactor stack. At the maximum tritium concentration of $0.75 \mu\text{c}/\text{cm}^3$ (Table 11.4-1) the secondary system would contain about 500 curies of tritium. A discharge rate of 20 gpm, or 10,000 lb/hr of steam, with at least one Graphite Reactor cooling air fan in operation would be used. The single fan operation gives an air flow up the stack of about 300,000 lb/hr, which would assure that the steam discharge would be well mixed with the air above the stack top. The power requirement for the evaporation is 11.1×10^6 Btu/hr for the 20 gpm liquid flow rate. At a continuous discharge of 20 gpm the secondary system would be emptied in 6.1 days.

Since the secondary discharge would be stopped during unfavorable atmospheric conditions, the total time to empty the secondary system might be 12 to 20 days.

At a discharge rate of 20 gpm, the tritium activity release rate is 0.95×10^{-3} c/sec. The maximum ground level concentrations for the four meteorological conditions discussed in Section 2.3.2 are given in Table 11.4-3. Even for the most unfavorable atmospheric condition, Case I, the maximum ground level concentration is only 5% of the AECM-0502 permissible level.

Table 11.4-3 Maximum Ground Level Tritium Concentration from Stack Discharge of Contaminated Secondary Water

Atmospheric Condition	Max. Ground Concentration ($\mu\text{c}/\text{cm}^3$ air)	Dist. of Max. Concentration (meters)
I. Weak neutral, inversion lid at stack top	9.5×10^{-9}	2600
II. Weak neutral	4.7×10^{-9}	2600
III. Strong inversion, typical night	3.4×10^{-10}	40,000
IV. Normal lapse, typical day	2.7×10^{-9}	600

Table 11.4-4 Maximum Ground Level Tritium Cloud Exposures and Doses from the Stack Discharge of Contaminated Secondary Water

Atmospheric Condition	Max. Cloud Exposure (c-sec/ m^3)	Maximum Dose (mrem)	Dist. of Max. Dose (meters)
I. Weak neutral, inversion lid at stack top	5.0×10^{-3}	0.66	2600
II. Weak neutral	2.5×10^{-3}	0.33	2600
III. Strong inversion, typical night	1.8×10^{-4}	0.024	40,000
IV. Normal lapse, typical day	1.4×10^{-3}	0.18	600

Since the discharge period would be at least 6 days long, the wind direction and atmospheric diffusion conditions would vary substantially and no area would be subject to the full tritium discharge burden. In setting upper limits to the total tritium doses which might be received from the secondary water discharge the atmospheric condition variation has been ignored, and the total doses have been calculated for each of the conditions of Section 2.3.2 on the basis that they are constant during the discharge period. The results of the total dose calculation are given in Table 11.4-4.

References

- (11.1) L. G. Epel, "Tritium, Carbon-14, and Argon-41 Dose Rates Due to HFBR Operations", BNL Memorandum (Feb. 20, 1964).
- (11.2) "Reporting and Investigating Accidents and Radiation Exposures", AECM-0502, Appendix, Annex 1 (April 4, 1962). The maximum permissible concentrations given in AECM-0502, Annex 1, are identical, for the radioactive materials of interest in this section, to the permissible concentrations given in "Standards for Protection Against Radiation", Title 10, Code of Federal Regulations, Part 20 (Oct. 15, 1962). These concentrations are less, in all cases, by a factor of 10 than the 168 hour per week occupational concentrations given in NBS-69 (see reference 11.3).
- (11.3) National Committee on Radiation Protection, Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure, U. S. Dept. of Commerce, National Bureau of Standards Handbook 69 (U. S. Gov't Printing Office, Washington, D.C., June 5, 1959), and Subcommittee 2, National Committee on Radiation Protection, "Addendum I to National Bureau of Standards Handbook 69", Health Physics, 9, 565 (1963).
- (11.4) Pinson and Langham, "Physiology and Toxicology of Tritium in Man", Journal of Applied Physiology, 10, 108 (Jan. 1957).
- (11.5) BNL Reactor Division experience in enriched fuel element procurement, also discussions with several fuel fabricators. Reported by J. A. Penney, Reactor Division.
- (11.6) L. G. Epel, "HFBR Stack Effluent Activity Due to Fuel Element Surface U-235 Contamination", BNL Memorandum (Jan. 29, 1964).
- (11.7) L. G. Epel, "Leak Rates and Radionuclide Concentrations in HFBR Cooling Water", BNL Memorandum (Dec. 19, 1963).
- (11.8) L. G. Epel, "Tritium Concentrations in Secondary Water and Doses from Cooling Tower Blow Down", BNL Memorandum (Feb. 27, 1964).

SECTION 12. INITIAL TESTS AND OPERATING PROCEDURES12.1 PREOPERATIONAL TESTS

12.1.1 GENERAL. The preoperational test period for the reactor starts when the completed plant is turned over to the BNL Reactor Operations Division. The preoperational test period ends with all process systems filled with the appropriate fluids, tested for proper operation, and with the plant ready for the initial loading of fuel. The steps necessary to accomplish this final condition are as follows. First, the various water-containing process systems must be cleaned and filled with deionized light water. The building utility systems must be placed in operation and properly adjusted. Each process system must be operated under a variety of conditions to test the proper operation of all components and instruments. After the test period, the entire plant is run with deionized light water in all water process systems for an extended period on three-shift, 24 hour operation. Finally, the heavy water systems are drained of light water, dried, and filled with heavy water, and various process tests repeated with heavy water.

12.1.2 PROCESS SYSTEM CLEANING AND FILLING. A large fraction of the HFBR plant consists of process systems containing light or heavy water. Each of these systems contains a variety of process components such as piping, valves, pumps, heat exchangers, filters, tanks, pressure vessels, an array of instrument connections, instrument primary elements, vents, drains, and relief valves. Each of these systems must be perfectly clean before the plant can be operated with nuclear fuel. The components and functions of the various process systems have been described in Sections 5, 6, 7, and 8.

The construction contractors responsible for the process systems are required to turn these over to Brookhaven in a clean condition. It is not possible to predict whether this will be carried out to the necessary degree of cleanliness. Since a period of preoperational testing with light water is necessary, it is intended to carry out further cleaning as needed during this period. Where additional cleaning of water process systems is deemed necessary the cleaning will be carried out along the following lines.

1. Pumps and exchangers will be protected with screens. Where practicable these components will be blinded off and bypassed or removed from the system. The reason for this is that these components are probably free of debris and can be visually inspected at the inlet and outlet connections. However, if a pump is removed from a system, another source of water under pressure will be needed. Such a source may be supplied by a temporarily installed auxiliary pump.
2. A flow of water sufficient to produce a velocity of 10-15 ft/sec will be used for flushing. The flow will be arranged to dump to a place where debris can be observed. The canal may be used for such ejected water and also as a supply source for the flushing water. Debris will accumulate in the deep pit for eventual removal.
3. Auxiliary connections to the various process systems may be needed. If so, these will be added.
4. Instruments will be disconnected during flushing. Instrument lines will be blown out with compressed air, blowing away from the instruments.
5. After flushing each system until it is cleaned, all valves in the system will be examined for action and tight shut-off. Bonnets may be removed for valve body inspection.

12.1.3 BUILDING UTILITIES. Building utility systems should be turned over to Brookhaven in operating condition. Such additional cleaning, flushing, and adjustment as may be necessary will be made in the domestic water, compressed air, electrical power, and fuel gas systems. The heating, ventilating, and air conditioning systems will have been started up and balanced by the construction contractors. The Reactor Operations Division will conduct flow measurements using a hot wire anemometer to see that the various ducts have proper flow. Measurements will be made to insure that the proper differential pressures are maintained for regions of lesser to greater potential contamination. Opening and closing of the air inlet and exhaust butterfly valves will be carried out and the automatic trip actions tested to insure that the closure is effective. The main exhaust fans will be tested for performance, maximum through-puts determined, proper motor operation checked, and automatic switching functions tried out. In order to determine the maximum capacity of the fans and the duct work system, the filters in the building exhaust filter bank will be removed and full flow tests made. The building vacuum breakers will be tested for proper operation, capacity, and reseating action. The building exhaust filters will be reinstalled and tested for leakage and general effectiveness by the DOP method.

The canal will be emptied and cleaned at an early stage in the preoperational period. The purification system BG-201 will be placed in operation, delivering deionized light water to the canal. The deionized water in the canal will be used as a supply source for flushing and cleaning operations.

12.1.4 PRIMARY SYSTEM. The primary system is of such a size that water velocities sufficient to flush it cannot be obtained without using the primary pumps. This system will be visually inspected for debris, vacuum cleaned, and generally wiped down to prepare for the introduction of deionized light water. Some sections of the system will be dismantled and opened to facilitate this inspection. The shutdown system is closely connected to the primary system and cannot easily be separated from it. For this reason the shutdown system is treated with the primary system. The heavy water storage tanks and transfer pumps and piping will be flushed out and placed in operation on deionized light water at the same time as the primary system. Similarly, the heavy water cleanup system will be flushed out and ion beds and filter media installed.

The primary system will be filled with deionized light water above the level of 20 in. P-101. This will permit the shutdown pumps and exchanger to be washed out. Flush water will be discharged into a temporary stainless steel screen basket inside the reactor vessel at the inlet from the shutdown system. The pump GA-105 will be started and the associated piping flushed out up to the helium eductor EG-101.

Dummy fuel elements having the same hydraulic characteristics as the regular elements will be installed in the reactor vessel and a temporary heavy duty screen located securely above the transition plate. Instruments on the primary system will be disconnected, the system closed and pressurized with nitrogen, and one main primary pump operated. The pump will be run at gradually increasing levels to wide open flow. The pump will then be shut down and the same procedure repeated for the other main pump. Then the reactor will be opened for inspection of the screen. After cleaning of the screen the system will be operated with both pumps wide open, then the pumps will be shut down and the screen reinspected. If debris continues to show at the screen the water charge will be discarded, taking advantage of this opportunity to blow out all drains of the heavy water system.

After satisfactory cleanliness of the primary system is achieved the

temporary screen will be removed and the control rod mechanisms and blades installed. Instrument lines will be blown out and reconnected. The overhead gas system on the reactor (helium system) will be blown clean with compressed air. The primary system will be refilled with clean, deionized light water, pressurized with helium, and the primary pumps started to establish normal flow. Shutdown pumps will also be started. Some cooling water will be needed on the exchangers to prevent the system from heating excessively. The temperature of the water will be held at about 125°F. An extensive set of preoperational tests will then be carried out, including (but not limited to) the items below.

1. A general inspection will be made for leaks, and the leak detection systems will be tested.
2. The head and motor current for each main pump will be measured as a function of flow rate.
3. The "wide open" capacity of the primary system will be determined for each of the main pumps operating alone and for the two pumps together.
4. The pressure loss at design flow will be measured for all components of the primary system.
5. All components and piping of the primary system will be checked for vibration at various flow levels from shutdown to well above the normal maximum flow rate. Particular attention will be paid to the fuel elements, control rod drive mechanisms, and the irradiation thimbles in the vessel.
6. The opening and closing times for all remotely operated valves will be determined. All valves will be operated at maximum speed at various primary flow rates to determine whether any water hammers occur.
7. Leakage from the pump and control rod mechanism shaft seals will be collected in containers during the preoperational test period to establish the leakage rates to be expected in subsequent operation.
8. With the primary pumps shut down, a test program of capacities, pressures, pressure losses, and valve closure effects will be carried out on the shutdown system.
9. A power failure to the main circulating pumps while operating at design flow will be simulated by opening the 2300 volt breakers, and the coastdown time for the primary system determined. In subsequent tests the main pump breakers will be opened with a shutdown pump running and the coastdown to shutdown flow measured.
10. The eductor pump GA-105 will be started to circulate water to the eductor. The eductor will circulate helium in the cover gas system. The gas pressure control valves PRCa-101 will be tested at various operating conditions. Gas will be deliberately injected to cause the discard system to function.
11. The shutdown pumps will be turned off to allow the flow reversal valves to open. The opening time for the flow reversal valves will be determined.
12. The ability of each transfer pump to restore liquid to the primary system at system pressure will be measured.
13. With the reactor liquid level at the normal point, DA drains will be opened to let the level fall to test the level alarm, scram, and low-low level safety systems.
14. Cover gas will be vented with the system at normal operating conditions to test the functioning of the pressure alarms, scram channels, and low-low pressure main pump interlock.
15. The effect on the primary system, operating at normal conditions, of a sudden opening of the siphon breaker valve HCe-101C will be

determined. Similarly, the effect on the primary system at normal operating conditions of the opening of the depressurization valve HCe-102 will be determined.

16. The control rod drive mechanisms will be tested with the reactor vessel both pressurized and at atmospheric pressure, and with and without full flow. Scram times will be compared for different drives and different operating conditions of the primary system.
17. The operation of the water cleanup system to maintain pH 5 in the primary coolant will be checked. Circulation will be maintained to the cleanup system during the preoperational test period to keep the light water at high resistivity.
18. Temperature instrumentation on the primary and shutdown pump systems will be tested by heating the primary coolant with the main pumps, and cooling it by secondary flow to the main exchangers.
19. With the poison water tank filled with deionized light water, the injection of liquid from the poison tank will be checked at full primary system pressure as well as with the primary system unpressurized.
20. The operation of the emergency electrical supply equipment will be checked, and all emergency systems will be operated on emergency power.

12.1.5 LIGHT WATER OPERATIONAL RUN. When tests of individual process systems have been completed, an integrated light water run of the plant of some duration will be undertaken. For this run all normal instrument interlocks which can be made operable in normal fashion without the presence of nuclear power and radiation will be placed in service. With all of the process systems running together, a general recheck of the capacities of components will be carried out and design conditions of flow, pressure, and temperature (where possible) will be established and held. Upsets will be introduced in all systems to determine the interactions which take place. Particular attention will be paid to the water purification systems to be sure that specified water chemistry conditions can be established and maintained.

The light water operation of the plant will be carried out on a 24 hour per day basis, to simulate the normal operating conditions of the reactor. The light water operation will continue until all operating personnel are thoroughly trained in the operation of all systems.

12.1.6 D₂O FILL. After the preoperation light water running period has established that all process systems and instruments are functioning satisfactorily, the plant will be shut down and preparations made for charging heavy water into the primary, shutdown, and auxiliary heavy water systems. The light water with which these systems are filled will be drained, and the various systems dried.

The filters and ion exchange media in the heavy water cleanup systems will be discarded and new filters and resin material installed in the beds. The resin beds will then be deuterated. Heavy water from the shipping drums will be charged into the storage tanks through the cleanup system. Charging of the primary system with heavy water will proceed after satisfactory water purity and pD conditions are achieved.

A brief series of process tests will be run with heavy water in the primary and shutdown systems to confirm the system characteristics extrapolated from the light water tests. After these process tests the plant will be ready for the loading of fuel.

12.2 INITIAL REACTOR LOADING AND CRITICALITY

This phase of startup operations begins when the preoperation checks have been finished. It ends when the reactor core is fully loaded, and low power tests are ready to start. The initial loading is carried out according to written procedures prepared specifically for the startup (see Section 12.6.5.12). The paragraphs which follow summarize the steps in the loading.

The highest reactor power needed for loading to criticality and for initial testing will be of the order of a few watts. Therefore cooling and pressurization are not needed, and the reactor vessel can be left open during these operations. Leaving the vessel open will require deactivating some of the interlocks and safety circuits that inhibit control rod use. These are discussed in Section 9.5.7. The nuclear and process instrumentation channels of the normal reactor system which will be operative during the initial loading are discussed in Sections 9.3.8 and 9.4.5.

The initial loading will contain no appreciable natural neutron source such as the photoneutrons that will be present after fission products have built up. A neutron source of approximately 10^7 n/sec will, therefore, be placed in one of the in-core thimbles. A number of detectors will be added to the normal reactor instrumentation during the initial loading and criticality tests, to extend the range of operation and to increase the redundancy. Three BF_3 counters or fission chambers and three boron-coated ion chambers will be placed around the core in beam tubes, where statistically significant signals are provided by the source. The display and electronics for these instruments will be in the control room. During the loading the counting rate of the scalers will be audible in the building over the paging system. The three ion chambers will be connected to the scram circuitry of the reactor in a one-out-of-three logic, with trip level at the full scale of the range setting in use. In addition, the nine neutron detectors used in the normal safety instrumentation of the reactor will be placed in the thermal irradiation thimbles to increase their sensitivity (see Section 9.3.8). These instruments provide period as well as power trips.

Before fuel loading begins, all fuel positions will be filled with dummy fuel elements and the source inserted in the core. Loading will proceed by replacing one dummy element at a time by a live element. During the loading, the heavy water will be at the normal shutdown level. While element changes are being made, the auxiliary control rod crown will be inserted and the main control rod crown will be partially withdrawn (cocked). The fuel elements to be loaded will be transferred one at a time from the top of the shield to the reactor pit. After each dummy element has been removed, the live element to replace it will be loaded.

Each fuel element will be loaded by being attached to the fuel handling lifting head and lowered into the proper position in the core. The rate of insertion of an element will be controlled by an Operator at the hoist control. The loading will be monitored by the Supervisor, who will observe and direct the operation from a position in the pit and will keep a record of the location of each element. After each element is loaded, a dust cover will be placed over the vessel opening, and the loading crew will leave the reactor pit.

The main rod crown will then be gradually withdrawn to the top of the core. The auxiliary crown will be withdrawn to the bottom of the core. The main and auxiliary crowns will be alternately withdrawn in small steps, a symmetrical pattern being maintained, while the change in multiplication is observed. If the reactor is still subcritical after full rod withdrawal, the

neutron density will be measured by the various detectors when the rods are full-out, when they are inserted to the core top and bottom in a symmetrical pattern, and when they are full-in. These points will be used for inverse multiplication plots. The next element in sequence will then be loaded by iteration of the previously described procedures.

After the reactor becomes critical on rod withdrawal, the critical positions of the main and auxiliary crowns in symmetrical pattern will be recorded following the loading of each additional element, to serve as guidance for the next loading. Multiplication with rods in symmetric positions at the core top and bottom and with all rods fully inserted will continue to be measured and plotted at each step until the core is fully loaded with 28 elements.

12.3 TESTS AT LOW POWER

This phase consists of the experiments needed for measurement of various nuclear properties of the reactor which are independent of the power level. Each experiment is carried out according to detailed written procedures (see Section 12.6.5.12). The sections below summarize the various tests and experiments.

During this series of tests, an effort will be made to keep the power as low as possible within the requirements of good signals from the neutron detectors, to avoid activation of the core and the resulting increased background.

12.3.1 CONTROL ROD CALIBRATIONS. A full study of rod shadowing effects was not possible in the critical experiments, in which only four main control rods were movable by mechanisms, while the other four main rods and all eight auxiliary rods were bolted in place. In order to establish that the reactivity effects of the complete set of control rods are as desired and to provide the operating staff with rod calibration curves, an extensive set of rod calibrations will be performed. These will be done with a reactivity computer of the kind used extensively during the critical experiments. This device has also been used to calibrate the control rods of the Brookhaven Graphite Reactor, the Brookhaven Medical Reactor, and reactors at other sites. It gives a time history of reactivity when the power level is within the dynamic range of the meter. The curve of control rod reactivity effect versus position is plotted with high accuracy as the control rod is moved, at as fast a rate as control rod drive motion allows (4.3 in./min).

A typical calibration consists of (1) taking the reactor critical at a power level of a few watts; (2) rearranging the control rods to increase or reduce the shadowing on the rod that has been selected for calibration, with this rod removed to the outer limit of the section to be calibrated; (3) stabilizing the reactor, to build up the delayed neutron source; (4) inserting the rod at normal speed until the power level falls below the dynamic range of the reactivity computer; and (5) returning to critical.

Due to the very low speed of the control rod mechanisms, complete plots of reactivity versus rod position cannot be made in one pass. The calibration will be made in sections. Calibration of rods in the main and auxiliary crowns will be made using the same procedure. In addition, the shutdown margin will be measured by total or partial scrams. Calibration will be done both in a shadowed configuration, and in a relatively unshadowed condition, where this is possible.

The performance of these measurements requires taking the reactor only to

a power of a few watts, and the rod motions required are similar to those used during startup, with the addition of some intentional scrams. The practices will be based on those used in critical assembly studies at Brookhaven.

12.3.2 FLOW EFFECTS. The effect of water flow on reactivity will be investigated by changing the water velocity and observing the resulting change of reactivity, if any. This test is meant to show if there is significant vibration or other motion of fuel elements or control blades due to hydrodynamic effects.

The reactor will be pressurized and brought to critical at a power of a few watts. Various flow rates will be established and the reactivity associated with any deviation at critical from a fiducial control rod configuration will be measured. Magnetic tape recordings of reactor noise will be made for subsequent analysis. These will be studied for existence of resonant departures from the zero flow condition.

12.3.3 UNIFORM TEMPERATURE COEFFICIENT OF REACTIVITY. The reactivity effects associated with uniform temperature changes will be investigated by heating the primary water with the pumping power of the primary pumps, with secondary coolant flow throttled. The primary and secondary flows will first be set to give a relatively low core temperature (limited by the dew point of the air around the vessel). The reactor will be brought to critical at a power level of a few watts. The secondary flow will then be gradually restricted to let the primary water temperature increase due to the pump heating. The temperature will be stabilized at various levels, and the control rod positions at critical will be recorded. From the previous control rod calibration, the curve of temperature defect of reactivity versus temperature will be obtained.

12.3.4 HELIUM BUBBLE EFFECTS. Experiments will be performed to establish the amount of reactivity held in helium bubbles (see Section 4.6.13). Helium pressure will be applied to the level needed to prevent cavitation in the primary circuit, and full primary flow will be established, with the secondary flow throttled to provide primary coolant temperature at or below the nominal operating value. The reactor will be taken critical at low power. The helium cover gas pressure will be increased in steps. The change in reactivity will be observed on a calibrated control rod or will be measured directly with the reactivity meter. At each stage of pressurization, a tape recording of the reactor neutron noise will be made, for observation of increased noise level from bubble population fluctuations. The rms noise will be measured, and the reactor transfer function determined.

Because the effects in question are small, there must be careful control of such variables as primary system temperature. It is possible that the search for helium bubble effects will be unsuccessful if the void fraction is much less than has been estimated, and if it is not possible to achieve very close temperature control. The experiment will establish an upper limit to the helium effect, even if the effect cannot be quantitatively measured.

12.4 TESTS AT INTERMEDIATE POWER

Power will be increased in steps, with careful assessment of operating performance being made at each stage. At the lowest power level at which reasonably reliable measurements of primary flow and of core inlet and outlet temperatures are available, a preliminary heat balance will be performed for initial calibration of power instruments. This calibration will be refined at an intermediate power (probably about 10 MW). At about 1 MW, the ion chamber positions will be fixed so as to provide the best range of ion chamber

currents for the nuclear instrumentation. At the various power levels, process instrument and equipment performance will be monitored and various special thermocouple and strain gauge records will be kept, as discussed below. Some physics tests will be performed. The intermediate power tests are carried out according to detailed written procedures (see Section 12.6.5.12). The sections which follow describe the tests briefly.

12.4.1 XENON TRANSIENTS. The reactivity transients produced by xenon-135 will be investigated by following transients after shutdown from equilibrium at various powers, and after operation at steady intermediate power for times of the order of the half-life of iodine-135. Typically, the reactor will be brought to a power level of the order of 10 MW (after a sufficiently long shutdown) and will be kept at this power for between 5 and 30 hours. Then the rods will be inserted to reduce the power to about 100 KW. The reactor will be held critical at this level for about 24 hours. The rod motion necessary to keep the reactor critical at the reduced power will be recorded, for interpretation in terms of reactivity.

The records of the transients will serve to refine the mathematical model describing xenon transients, and will provide guidance for the operation of the reactor. That is, the experiment will supply empirical constants to be used in the equation describing the buildup, decay, and burnout of the xenon, and the associated reactivity effects.

12.4.2 FLUX MEASUREMENTS AT EXPERIMENTAL FACILITIES. Neutron and gamma ray fluxes at the inner end and the exit end of beam tubes will be measured. Output neutron and gamma ray fluxes will be measured with a standard collimator placed in various beam tubes. A simple slow chopper will be used to measure the spectrum of neutrons from various beam tubes. The reduction of beam intensities by shutter insertion will be measured. These experiments will indicate to what extent the initial design goals have been met, and they will assist in final guidance for installation of the experimental equipment at the reactor.

12.4.3 REACTOR VESSEL TEMPERATURE AND STRESS. The lower portion of the reactor vessel is covered with a network of 48 thermocouples and 36 strain gauges. These thermocouples and strain gauges are all attached to the outer surface of the reactor vessel. Connecting wires from the thermocouples and strain gauges pass upward along the vessel neck into the upper pit, and are taken out of the pile top through horizontal access holes in the shield. Temporary instrument setups for temperature recorders and strain gauge measurements will be made on the operations floor beside the pile top. The vessel temperature and strain gauge instrumentation is not considered part of the normal reactor instrumentation. The thermocouples and strain gauges will have a limited life in the high radiation field of the lower reactor cavity, and these instruments are considered useful for the startup and initial full power operating periods only.

All of the thermocouples attached to the reactor vessel are Chromel-Alumel couples. Twelve of the 48 thermocouples form a network which gives the general temperature distribution in the spherical section of the reactor vessel. These thermocouples are located away from the nozzles of beam tubes to avoid the local temperature variations which may occur in those locations. The 12 couples are located to give an equatorial temperature distribution for the sphere and a vertical temperature traverse starting from the bottom of the vessel on the vertical centerline and proceeding up the vessel wall to the cylindrical neck above the sphere. The vertical traverse is at azimuth 043°, midway between beam holes H-1 and H-2. These 12 thermocouples are of Chromel-Alumel wire insulated by hard-packed magnesium oxide powder in type 304 stainless steel

sheaths. The attachment of the couples to the vessel surface is made by shrink-fitting an aluminum tube over the end of the stainless sheath. The aluminum cover and stainless sheath are then flattened at the sensing end and are welded to a 1/8 in. thick aluminum plate which is spot welded to the vessel.

The remaining 36 thermocouples are also Chromel-Alumel couples but are not protected by stainless steel sheaths. The set of 36 couples is part of the strain gauge instrumentation, with one couple located at each strain gauge location.

The strain gauges are located for the most part at points of high stress in the nozzles of the beam tubes. The beam tubes H-4 and H-7 are instrumented because they are the thimbles with the greatest angles from the normals to the spherical vessel surface. Beam tube H-9 is also instrumented because of its large size and high heat rate. The 36 strain gauges are located in pairs at points around the nozzle entrances for the tubes H-4, H-7, and H-9, with two additional gauges at the transition from the spherical section of the vessel to the cylinder. One strain gauge is located at the midplane of the spherical section of the vessel, away from beam tubes, for comparison purposes.

The 36 strain gauges, and the 36 Chromel-Alumel thermocouples associated with them, are attached to the surface of the reactor vessel by an aluminum oxide deposition method developed by the Aerojet-General Corporation of Azusa, Cal. The Aerojet-General Corporation performed the task of attaching the thermocouples and strain gauges to the vessel at the reactor site at BNL.

Strain gauge readings will be taken during the startup period of the reactor. Zero point measurements will be taken with no liquid or pressure in the reactor vessel. Subsequent measurements will be made when the vessel is filled with D_2O , when the system is pressurized at various levels to full design pressure, with the system at full flow and design pressure, and finally at various stages of nuclear power as the reactor is started up. The results of the strain gauge measurements with and without nuclear power will provide a check on the analytical work done in the design of the vessel and will indicate what safety factors exist at normal operating conditions. The measurements will also provide a basis for extrapolation of vessel stresses at higher power levels than the design 40 MW.

Temperature readings from all thermocouples will be made at various stages in the nuclear power escalation. As with the strain gauge measurements, the temperature measurements will provide a check on the design calculations, and will provide a basis for extrapolating the vessel temperatures at higher nuclear power levels than the design 40 MW.

In addition to the 48 thermocouples and 36 strain gauges described above, spring-loaded Chromel-Alumel thermocouples will be mounted in several of the high heat-rate beam tubes. These thermocouples will be inserted on long extension rods working through special shield plugs from the experimental level. Temperature profiles in the regions of the beam tube ends and finned sections will be made by this means at various powers during startup.

The vessel temperature and strain gauge measurements will be made at each power level as the nuclear power of the reactor is increased. The temperature and strain gauge runs require that the reactor power be constant for a long enough time for all temperatures to come to equilibrium in the reactor vessel. This equilibrium is achieved, even at full reactor power, in a relatively few minutes. The run must then continue for a long enough time to allow readings to be made on all thermocouples and strain gauges.

12.4.4 SHIELD TEMPERATURES. There are 52 thermocouples mounted in the thermal and biological shields. All of these thermocouples are Chromel Alumel ungrounded junctions insulated with hard-packed magnesium oxide powder in type 304 stainless steel sheaths. These thermocouples are not considered part of the normal reactor instrumentation. Those thermocouples which are close to the inside of the reactor cavity will have a limited life in the high radiation field.

Two thermocouples are attached to the boral sheets which are banded to the reactor vessel. One of these couples is at azimuth 000° , immediately below the primary coolant outlet nozzle, and the other is at the same elevation at azimuth 180° . The connecting wires from these couples are carried up into the reactor pit.

Four thermocouples are located on the upper thermal shield. Three of these are at azimuth 043° , with one each at radius 30 in. on the upper and lower shield surfaces, and the third on the upper surface of the support ring at radius 42 in. The fourth couple is at azimuth 317° on the lower surface of the shield at radius 30 in. The thermocouple connecting wires are carried up into the reactor pit.

Twenty-four thermocouples are located on the inner and outer walls of the lower thermal shield. Ten of these couples are at azimuth 043° and run on both inner and outer wall surfaces from the bottom of the shield at the vertical centerline to the upper support ring. The other fourteen couples are arranged in two vertical traverses on the inner and outer walls at azimuths 165° (8 couples) and 317° (6 couples). The lead wires from thermocouples on the inside surface of the lower thermal shield are run up to the reactor pit. The lead wires from the couples on the outside surface of the shield are run out through the heavy concrete to the shield face on the experimental level.

Four thermocouples are located in vertical wells in the lead layer of the thermal shield. The wells are accessible from the equipment cells beneath the reactor. These four thermocouples are the only couples in the shield group which are replaceable. Connecting wires are carried down into the equipment cells.

Eighteen thermocouples are imbedded in the heavy concrete of the biological shield. These are arranged to form vertical and radial traverses in the shield annulus to permit temperature distributions to be determined. The lead wires from these couples are run to terminal heads at the shield face on the experimental level.

As with the reactor vessel temperatures, readings will be made of the shield temperatures at various power levels during the power escalation period. The temperature distributions in the thermal and biological shields, as determined from the thermocouple measurements, will be used as a check on the design calculations for these components, and will provide a basis for extrapolating the shield temperatures to higher nuclear power levels than the design 40 MW. The detailed temperature distributions from the thermocouple measurements will also be used to adjust the cooling water flow to the various components of the thermal shield and to the cooling coils in the concrete.

12.4.5 SHIELDING MEASUREMENTS. Radiation level measurements will be made around the outside of the reactor shield and in the reactor pit area at low nuclear power levels to determine where additional fixed shielding is necessary. It is likely some additional shield plugs and blocks will be necessary at the seal plate floor level in the reactor pit. As shield weaknesses are corrected,

radiation level measurements will be repeated at successively higher nuclear power levels during the power escalation period.

Radiation levels will be measured at the locations of the primary system pumps, valves, and principal pipe gaskets. Radiation levels at the control rod mechanism external drive components and at the shutter drive motors will also be measured. These measurements will indicate the life to be expected from various insulating and packing materials which are subject to radiation damage.

Radiation levels will be measured around the reactor pile top, in particular around the removable shield plug sections and the periscope shield blocks.

12.5 TESTS AT FULL POWER

Most of the testing to be done at full power is simply an extension of the tests mentioned in the section above, carried out at full nuclear power of the plant. The additional tests which become possible at full power involve running for several fuel cycles to confirm the proper operation of the plant and to obtain a final experimental check on the core burn-up and fission product reactivity requirement.

12.6 OPERATING PROCEDURES

The reactor is operated according to detailed written procedures. These documents are listed and described in Section 12.6.5. Sections 12.6.1 through 12.6.4 give brief descriptions of the operation of the plant in the normal cycle.

12.6.1 NORMAL OPERATION CYCLE. In normal operation the reactor is operated continuously at 40 MW power for 19 days. The reactor is then shut down for 2 days to allow fuel to be changed, experimental equipment to be adjusted, and maintenance work to be done on the system. After the 2 day shutdown the reactor is started up and runs continuously for 19 days again at 40 MW power. In the sections below, the operating sequence is outlined, starting just before a normal startup.

12.6.2 PRESTARTUP CONDITIONS. This section summarizes the sequence of operations which are carried out prior to startup. At this time the core loading is 28 elements, of which the outer 14 have been repositioned after being partially burned in the previous cycle, and the inner 14 are fresh fuel elements. The control rods are fully inserted. Cooling flow is maintained at 800 gpm through the core by a shutdown pump.

In preparation for the startup the vessel is carefully inspected for debris, loose tools or bolts, and for the proper position of all fuel elements and control rod blades. The flow reversal valves are tested by shutting off the driving head supplied to them from the shutdown pump which is running. After the test the valves are closed by re-establishing the driving head. The reactor vessel is then closed and the top cover bolted down. The D₂O level in the vessel is raised to the maximum height to drive as much air as possible from the gas space. The gas space is then purged with helium to establish the normal cover gas atmosphere. The D₂O level is readjusted to the normal cold level. Helium inlet flow is established and the pressure controller is set for 200 psig. Pressurization of the system then follows as the helium pressure builds up. The vessel is checked for tightness at the top closure. The "run" pumps in the experimental facilities cooling system, the thermal shield system, and the biological shield system are turned on and normal flows and liquid levels established in the auxiliary systems.

The primary pumps are then started, one at a time, and the primary system brought to normal flow and differential pressure. Normal circulation rate in the secondary system is established. The shutdown pump which had been running during the shutdown continues to run, although the flow in the shutdown system is reduced to the pump cooling bypass flow after the main pumps are started (see Section 7.1.1).

The reactor is now ready for startup. In addition to the steps outlined above in preparing the primary and auxiliary systems for startup, the Operators have checked to see that all shield plugs, irradiation experiment casks, and shields around beam experiments are in place for the coming run. The building exhaust system has been checked and is operating normally and the building containment shell is tight. All instruments show normal temperatures, flows, liquid levels, and pressures. The nuclear safety system and startup instrumentation has been tested and calibrated during the shut down. The nuclear instrument channels show "on scale" readings due to the photoneutron production in the heavy water from the fission products present in the outer 14 elements of the core. These "on scale" readings should be apparent on the linear power channels even with control rods fully inserted. The excess radioactivity of the core if all rods were fully withdrawn would be 10.6 %k.

12.6.3 STARTUP AND OPERATION AT NORMAL POWER LEVEL. To start the reactor the Operator, under the direction of the Supervisor, turns on the key switch and activates the main control rod clutches. He withdraws the main rod crown to the expected critical position. For a normal startup the critical position for the main crown will be between rod position 13 and 14 in., i.e., with the tips of the main rods 1/8 to 1-1/8 in. below the top of the active core.

The auxiliary rod crowns is then withdrawn in small steps by the Operator until indications of a positive period are shown on the appropriate meters and on the safety channel trip meters. The position of the auxiliary rod crown should be between rod positions 13 and 14 in., symmetrical with the main rod crown about the core center, at criticality.

The Supervisor remains in the control room as the Operator controls the rate of power rise to a suitable value by observing the "percent of trip" meters on the nuclear safety channels. As the reactor power approaches the desired operating power the Operator increases the period to infinity with the auxiliary rod crown and then switches to automatic power level control.

The automatic control operates the single auxiliary rod, A-1. The control system moves rod A-1 in and out to maintain a constant flux at the control channel ion chambers. As the fuel burns up, xenon grows in, and irradiation samples are placed in or removed from the reactor, the automatic rod will move in and out. However, the net motion will be outward as the fuel burns up. When the automatic rod has made a net outward motion of 1 in., a signal is given to the Operator. The Operator then withdraws one of the other 15 rods about 1 in., causing the automatic rod to return to its initial position and start moving again to maintain constant flux at the ion chambers. Repetition of this process for each of the other 15 rods results in a net withdrawal of the main and auxiliary crowns of 1 in. The alarm range for the automatic rod is then reset to conform to the new crown positions, i.e., one inch further out. Again the automatic rod works its way outward 1 in. and is returned to the new starting position by movement of one of the other 15 rods. The process continues until all rods end near the full-out positions at the end of the cycle.

During the startup the Supervisor remains in the control room where he can maintain proper surveillance of reactor conditions. After the reactor is operating on automatic control at the normal power level, the Supervisor inspects the plant and directs the Operators as needed. One Operator remains at the console at all times while the reactor is operating. The Console Operator keeps records of power level, rod position, and flow and temperature conditions. Other Operators check operating process equipment in all areas of the plant, recording operating data on appropriate logsheets.

The liquid level in the reactor vessel will approach the low level alarm point several times in each operating cycle, due to the continuous bleeding of primary heavy water to the storage tanks through the fission product water monitor. Normally the level will be corrected before the low alarm sounds. To correct the low level condition an Operator is dispatched to the equipment level where he establishes telephone communication with the console Operator. The roving Operator starts a transfer pump (see Section 7.6) and admits heavy water to the reactor vessel from the storage tanks FA-101 or FA-102 until the console Operator notifies him that the proper level has been reached.

During cold weather, the roving Operator may be dispatched to the cooling tower area to shut off fans, open bypass valves, or shut down a tower cell to maintain the desired 65°F minimum temperature of the secondary coolant. Similarly, in hot weather, the reverse operations will have to be carried out to maintain the desired 85°F maximum temperature.

12.6.4 SHUTDOWN PROCEDURES. The normal shutdown is accomplished either by running in the rods at normal speed or by deliberately tripping a 2 out of 3 scram or setback instrument channel to test the safety system. After the rods are fully inserted, the console is locked down. The primary system is maintained at full flow for an additional 10 minutes to insure that system temperatures have reached the new equilibrium values. The main pumps are then shut down and the secondary flow to the main exchangers cut off. The shutdown pump maintains the necessary cooling flow to the reactor core.

The vent valves at the secondary inlet lines to the main exchangers are opened, reducing the water pressure on the exchanger secondary sides to just the static head in the shells. The primary system may be depressurized after the exchanger secondary inlets are vented. A few pounds gauge pressure of helium are retained in the cover gas space if the system is depressurized before fuel changing is ready to start. The cooling flows in the auxiliary systems are decreased or shut off, as appropriate.

After 12 hours the fission product decay heat is less than 200 KW and the sodium activity in the heavy water has decayed to a level which will allow operations personnel access to the head of the reactor vessel. Prior to opening the reactor vessel the heavy water level is raised to the maximum elevation and the helium blanket vented to eliminate any residual volatile fission products.

The reactor vessel fuel handling cover is unbolted and removed. Air exhaust from the vessel neck via 2 in. P118 to the off-gas manifold is set up to reduce the diffusion of tritium-bearing heavy water vapor into the pit area. The fuel handling equipment is set in place over the vessel and tested. The details of the fuel handling equipment and procedures are given in Section 7.8.

During fuel handling the console is locked down, with an Operator in attendance and the neutron sensing channels in operation. The control rods are fully inserted. One shutdown pump is operating, delivering 800 gpm to the

12.6.5 OPERATING INSTRUCTIONS AND RECORDS. The reactor is operated according to written instructions. The primary source of these instructions is the HFBR Operating Procedures Manual. Supplementary instructions are given in a Temporary Operating Procedures Logbook and in a Shift Instruction Logbook. Records of reactor operation are kept in a Shift Operations Logbook, in Console and Process System Shift Logsheets, on chart records from various instrument recording devices, in a Canal Logbook, and in a Fuel Movements Logbook. Fuel Charging and Discharging Record Sheets, and the Tickler File Card System constitute both written instructions for certain procedures and a record of the operations carried out. Maintenance work is carried on according to the Reactor Maintenance Manual. These instruction and record books are discussed in the sections which follow.

12.6.5.1 HFBR Operating Procedure Manual. The HFBR Operating Procedures Manual is the governing document with regard to operation of the reactor. The Manual is prepared by the Reactor Operations Division. The Manual is approved by the Head, Operations Division, and by the Deputy Director after review by the Reactor and Critical Experiments Safety Committee. All phases of HFBR operations are prescribed in detail in the Procedures Manual, including all operating limits and trip points.

Changes in the Operating Procedures Manual may be made only with the approval of the Head, Operations Division, and also with the approval of the Deputy Director after review by the Reactor and Critical Experiments Safety Committee for any procedure or operating limit which affects nuclear safety. Copies of the HFBR Operating Procedures Manual are kept on file with the Reactor and Critical Experiments Safety Committee and with the AEC Brookhaven Office.

The HFBR Operating Procedures Manual has been prepared in a draft form similar to that used for the other BNL reactors. The details which are required in the Procedures Manual for complete coverage of all necessary operations require actual experience by the operating staff with the reactor equipment. This experience can be achieved only after the construction contractors have turned the plant over to BNL. The initial period of operating the plant equipment with light water and without fuel will provide the necessary experience and familiarity with the equipment for a final review of the draft Operating Procedures Manual. As a result of this review, such modifications in procedures as seem prudent and necessary will be made. Operation of the reactor with nuclear fuel will not be carried out until such modifications have been made in the Operating Procedures Manual and the Manual has been reviewed by the Reactor and Critical Experiments Safety Committee and approved by the Deputy Director. The Brookhaven Office, AEC, will participate in this review, at least to the extent of attending the meetings of the Reactor and Critical Experiments Safety Committee.

12.6.5.2 Temporary Operating Procedures Logbook. The Temporary Operating Procedures Logbook for the HFBR is maintained by the HFBR Operations Group Leader. This logbook contains instructions and procedures for the operation of the reactor and associated facilities which are expected to be in effect for more than a few shifts and less than a few months. The temporary operating procedures do not apply to reactor operating limits or to rules or procedures which affect nuclear safety, unless the Head, Operations Division, and the Deputy Director, after review by the Reactor and Critical Experiments Safety Committee, have approved the change.

12.6.5.3 Shift Instruction Logbook. The Shift Instruction Logbook contains specific instructions for duties to be carried out at a given shift. These matters may apply to preventive maintenance work, experimental sample handling, and similar matters which are not connected with reactor operating limits or nuclear safety. A number of people in the Operations Division are authorized to place instructions in the Shift Instruction Logbook. A list of such persons is kept in the logbook. The list includes the Operations Group Leader, the Supervisors, the Research Coordination Group, and the Maintenance Group.

12.6.5.4 Reactor Maintenance Manual. The Reactor Maintenance Manual contains specific instructions for maintaining the various reactor equipment. Procedures associated with maintenance activities which are designed to insure that the reactor is in a safe condition during the period of maintenance are contained in the HFBR Operating Procedures Manual.

12.6.5.5 Tickler Card File System. The Tickler Card File System is maintained by the Operations Clerk for the HFBR. The card system is used to remind operations division personnel of recurring requirements, particularly in the areas of maintenance and inspection. Tickler cards issued to Supervisors for maintenance and inspection work contain both instructions as to what is to be done and a place to record the completion of the job.

12.6.5.6 Fuel Charging and Discharging Record Sheets. The Fuel Charging and Discharging Record Sheets are issued by the Operations Group Leader prior to each shutdown. The Fuel Charging and Discharging Record Sheets contain explicit instructions for the movement of used fuel out of the reactor and new fuel into the reactor. The sheets also provide a place for the Supervisors to record specific fuel movements made at the shutdown.

12.6.5.7 Shift Operations Logbook. The Shift Operations Logbook is maintained by the Supervisors. It provides a record of the operation of the reactor, shift by shift. The Shift Operations Logbook entries include data on operating conditions at the beginning of each shift, an outline of major reactor conditions and shutdown work done, important occurrences affecting operating conditions or the safety of personnel or equipment, all items needing maintenance or repair, and experimental sample irradiations made during the shift.

12.6.5.8 Console and Process System Log Sheets. The Console and Process System Log Sheets are operating records filled in by the Operators during each shift. The Console Log Sheet includes a record of reactor power level and core reactivity as a function of time, and also calculated heat balances to compare to the power calculator readings. The Process System Log Sheets contain a record of liquid levels, flow rates, pressures, differential pressures, temperatures, conductivities, radiation levels, and any unusual conditions in all process systems and utility systems throughout the plant.

12.6.5.9 Recorder Charts. Recorder charts from the control room instrumentation are kept on file for a period of time to provide detailed operating records of all process systems. The recorder chart files supplement and provide a cross check on the Process System Log Sheets maintained by the operating crews.

12.6.5.10 Canal Logbook. The Canal Logbook is maintained by the Canal Area Supervisor. He records in the Canal Logbook a summary of canal operations, including the shifting of freshly discharged fuel from the temporary storage rack to the cutoff saw station, and subsequently to the storage baskets.

12.6.5.11 Fuel Movements Logbook. The Fuel Movements Logbook is kept by the Canal Area Supervisor. The Fuel Movements Logbook provides a complete record of the location of all fuel elements stored in the canal.

12.6.5.12 HFBR Startup Procedures. The procedures required for the startup and initial evaluation of the plant are separate documents, but will, to the greatest extent possible, utilize the standard procedures for equipment operation. These procedures will be prepared by the Reactor Startup and Evaluation Team, consisting of senior members of the HFBR Project Group and of the Experimental and Theoretical Reactor Physics Divisions. This team will act in an advisory capacity to the Reactor Operations Division.

The Reactor Division will review the Startup Procedures for consistency with the capabilities of the installed equipment. The Startup Procedures will be approved by the Head, Operations Divisions, and by the Deputy Director, after review by the Reactor and Critical Experiments Safety Committee.

The initial operation of the plant will be physically conducted by qualified Operators and Supervisors, with advice from members of the Reactor Startup and Evaluation Team. The responsibility for carrying out the operating steps in accordance with the Startup Procedures, or any approved modifications thereof, is solely that of the Reactor Division.

12.7 MAINTENANCE PROCEDURES

12.7.1 GENERAL. Maintenance procedures on all portions of the reactor plant are detailed in the Reactor Maintenance Manual. Operational control of maintenance activities is outlined in Section 13. Operating procedures connected with maintenance activities are given in the HFBR Operating Procedures Manual.

Most of the equipment in the reactor plant is of a conventional nature and no special tools are required for maintenance. The components within the reactor vessel do, however, require special handling tools. These have been provided as part of the reactor equipment. The special tools are, for the most part, long-handled torque wrenches and clamping devices.

Maintenance work which involves opening a process system which contains D₂O is carried out with the equipment shrouded by temporary hoods or plastic tents. The hoods or tents are exhausted to the building exhaust system to prevent the diffusion of tritium-bearing heavy water vapor into the general building atmosphere. Personnel carrying out maintenance tasks on heavy water systems equipment wear protective clothing, including polyvinylchloride suits which are supplied with breathing air. The surveying of equipment contamination and of radiation exposures during maintenance work is done by the Health Physics Group.

12.7.2 MAIN PUMPS AND MOTORS. The main pump motors should require little maintenance beyond periodic oil changes. Pump motor lubricating oil will be changed once each year. During the first year of operation, samples of the oil will be taken every two months to determine whether any radiation effects are occurring to reduce the effectiveness of the lubrication. Major maintenance on pump motors will require dismounting the motors and removing them from the equipment cells to the manufacturer's plant.

The primary pumps will require periodic maintenance, particularly on the shaft seals. The main pump seals may be replaced without dismounting the pump motor. Replacement of wearing rings, pump shafts, or impellers will require

dismounting the motors. Maintenance work on the pumps requires that the heavy water in the associated process lines be drained.

12.7.3 PRIMARY VALVES. Maintenance work on the primary valves which involves opening the system will require that D_2O in the connecting pipe lines be drained. Maintenance work and lubrication of the valve operators will not require that the system be drained. A spare operator has been purchased for each type of valve so that a malfunctioning operator may be replaced quickly and removed from the cell area for repairs. Should dismounting of the valve bodies be required, the attached lines will be drained, the valve removed from the cell area, and decontaminated before repair work starts.

12.7.4 PRIMARY COOLERS. Maintenance work on the primary coolers involves draining both the D_2O and H_2O sides. Working space in the shielded cell area around the coolers is very limited. Should it become necessary to open the coolers to plug leaking tubes, the stacked block shield wall at the end of the cell will be removed, the steel gas plate removed, and the exchanger moved halfway out of the cell to obtain working room.

12.7.5 SHUTDOWN SYSTEM. The shutdown pumps and motors are located on the equipment level outside the shielded cell area. Maintenance work on the pumps themselves will require draining the associated piping and erecting a plastic tent around the pump to be worked on. Shaft seal maintenance is the principal expected activity for the shutdown pumps. The pump motors are accessible above the local shields and are easily dismantled for whatever maintenance work is necessary. The shutdown cooler is located in the north shielded cell where working room is quite limited. Maintenance work on the cooler of any extensive nature will require disconnecting the cooler and removing it from the cell. Draining of the system is necessary for this operation.

12.7.6 D_2O PURIFICATION SYSTEM. The D_2O purification system vessels are designed so that they may be valved off, disconnected, and drawn up into a shield by the overhead monorail hoist. Exhausted resin beds and plugged filter units will be removed to the laboratory "hot" storage area to allow decay of radioactive material. Spare bed and filter vessels have been provided with the reactor equipment.

12.7.7 AUXILIARY PROCESS SYSTEMS. Maintenance of the auxiliary process systems will be concerned principally with pump shaft seals, pump motor bearings, lubrication, and occasional checks of coolers for leaking tubes. The equipment is all of conventional nature and the maintenance procedures are quite straightforward. In all systems due precaution must be taken for radioactive contamination.

12.7.8 CONTROL ROD MECHANISMS. Preventive maintenance on the control rod drive mechanisms is carried out at each shutdown by checking the operation of the drive units and lubricating the external drive assemblies. The external drive assemblies are made up of a number of subassemblies, and component repair work is expedited by substituting spare subassemblies and removing the faulty units to the machine shop. Normal maintenance requirements for the external drive package include the resetting of limit switches each time the internal mechanism is readjusted, and the occasional replacement of a shaft seal.

After the initial installation and adjustment, the internal portion of the control rod mechanisms should not require much servicing. The control rod blades will be changed at intervals of about two years. Major maintenance work on an internal mechanism requires the substitution of a spare internal drive and the removal of the faulty unit to the shops for disassembly.

SECTION 13. ADMINISTRATION13.1 OPERATING ORGANIZATION

13.1.1 OPERATIONS RESPONSIBILITY. The operations and safety of the High Flux Beam Reactor are line responsibilities of the Director of Brookhaven National Laboratory and the Head of the Reactor Operations Division. The organization of the laboratory and of the Reactor Division for the operation of reactors, including the HFBR, is shown in Table 13.1-1.

The Director has delegated to the Deputy Director responsibility for all matters pertaining to the reactors operated by the laboratory. The Head, Reactor Operations Division, reports to the Deputy Director, and communications from the laboratory to the AEC on matters connected with all phases of reactor operation are signed by the Deputy Director. In addition to the reports made by the Head, Reactor Operations Division, the Deputy Director receives further assurances of nuclear, radiation, and industrial safety through the actions of other organizations reporting directly and independently to him. These are listed in Section 13.4.

The responsibilities of the Reactor Division in matters not directly concerning the HFBR are indicated in Table 13.1-1 and are not discussed further.

The assignment of responsibilities within the Reactor Division for operation and maintenance of the HFBR are given in this section and those which follow. These assignments are designed to meet the requirements set forth in the Prime Contract (AEC Contract No. AT-30-2-GEN-16), AEC Manual Chapters 8401 and 0502, and to provide for the efficient and safe operation of the plant.

All Group Leaders within the Reactor Division report to the Division Head. However, the Instrument Maintenance, General Maintenance, and Water Chemistry Group Leaders report directly to the HFBR Operations Group Leader on matters pertaining to direct and routine service to the HFBR. The Group Leaders do not have the authority to initiate a change in design to the reactor facility, although in most cases the Group Leaders participate in the design and report preparation incidental for processing a change. The authority for all changes in design, procedures, and limits comes from the Division head. The Division Head, in turn, is responsible for obtaining the necessary recommendations, approvals, and concurrences in these matters as may be required from the safety committees, the Deputy Director, and the AEC.

The Division Head has the responsibility of keeping the Deputy Director and the Reactor and Critical Experiments Safety Committee informed of the condition of the HFBR. This responsibility is largely fulfilled in the Monthly Report of the Reactor Division and by other reports listed in Section 13.2.5.

13.1.2 HFBR OPERATIONS GROUP

13.1.2.1 Group Leader. The HFBR Operations Group Leader reports directly to the Reactor Division Head and is assigned the responsibility for maintaining continuous operation of the reactor in accordance with the approved procedures and within the approved limits. Whenever the approved procedures become inadequate or whenever there is no written procedure to cover the mode of operation desired, the Group Leader has the responsibility of preparing the procedure for appropriate review and approval.

Table 13.1-1 Reactor Division Organization

The HFBR Operations Group Leader schedules the routine operation of the reactor and assigns qualified Reactor Supervisors and Operators to meet routine reactor operating requirements. He also assigns qualified Reactor Supervisors and Operators to perform miscellaneous duties such as spent fuel handling, fuel testing, special inspections, etc. All fuel movements in the reactor are authorized in writing by HFBR Operations Group Leader.

13.1.2.2 Shift Supervisors. The responsibilities and authorizations assigned by the HFBR Operations Group Leader to the Shift Supervisors are limited and specific. The Shift Supervisors are responsible for seeing that all of the work done on the reactor is in accordance with the approved procedures and within approved limits. Each Supervisor and Operator's work schedule is specific as to date and time with the added specification that their responsibility continues until they have been properly relieved by those assigned. Supervisors have the authority to call in qualified Operators in addition to those assigned. They also have the authority and specifically assigned responsibility to shut down the reactor whenever they think it is not being operated safely. Supervisors do not have the authority to alter procedures or limits. They do have the authority to shut down whenever, in their opinion, the procedures or limits become inadequate.

Shift Supervisors are also responsible for the actions of the Operators assigned to them and provide the supervision necessary to assure that the Operators comply with approved procedures, and meet training requirements and quality standards.

13.1.2.3 Plant Operating Crews and Supervisors. The plant has been designed to operate on a continuous basis with a minimum of two Operators in attendance. Until this can be confirmed during operation, each operating shift will consist of one qualified Reactor Supervisor and three qualified Reactor Operators. In the future, with successful experience in routine operation as further justification, it is expected that one Supervisor will be able to supervise both the HFBR and the BGRR when they are in routine, constant power level operation, and that the operating crews of the two reactors will be integrated and adjusted to the extent that is practical and consistent with safe and efficient operation. This integration is made possible by the close proximity of the two reactors and by the familiarity of the operating personnel with both reactors.

There are certain important minimum requirements which must be achieved in routine operation as well as during shutdown and maintenance periods. These are covered in detail in the HFBR Operating Procedures Manual. A summary follows.

- (1) A qualified Supervisor will be in direct charge of all operations and maintenance of the reactor and will be responsible for assuring that operation and maintenance is in accordance with approved procedures.
- (2) A qualified Reactor Operator will be at the reactor console when the reactor is being fueled or whenever the control rods are in a condition to be moved. When fuel movements are being made, the console will be locked, protecting against any control rod movement, and the counting rate and period instruments will be operating.
- (3) No one other than those specifically qualified to do so will operate any equipment on or about the reactor which affects reactivity. This does not, of course, preclude the training of persons under the

immediate direction of qualified Operators.

- (4) During each startup, a qualified Reactor Supervisor will be in the control room until constant power operation is achieved. During this period, the Supervisor will be unavailable for duties not directly related to the startup of the HFBR.
- (5) The assistance of the Health Physics Organization is required in the monitoring of any in-reactor work. This includes all work inside the biological shield, fuel handling, control rod replacement, work within the primary process cells, and the removal of collimators and other irradiated items.

13.1.3 WATER CHEMISTRY GROUP. The Water Chemistry Group will consist initially of one Chemist and one Technician. It will be the responsibility of this group to maintain water and gas quality within the desired standards in the primary and secondary (i.e., D_2O and H_2O) systems of the HFBR. This group, with the assistance of the Operators, will sample, analyze, and report conditions of the systems. The assistance of the Hot Chemistry Laboratory in the Nuclear Engineering Department will be required for some analytical chemistry and the assistance of the Research Coordination Group will be required for gamma ray analyses.

The Water Chemistry Group will replace ion exchange media, regenerate water treatment systems, and deuterate and dedeuterate resin columns as necessary. Water and gas quality conditions will be reported in the Monthly Report of the HFBR.

13.1.4 RESEARCH COORDINATION GROUP. The Leader of the Research Coordination Group is assigned the general responsibility of working with the experimenters to assure that the reactor facilities are used by them to the best advantage, and at the same time are not used in a way which will jeopardize the safety of the reactor. It is for this reason that a representative of this group installs and removes experimental devices.

With regard to beam experiments, the experimenters have a wealth of reactor experience. It will, however, be necessary for the Research Coordination Group to review collimator designs to see that the containment criteria is not violated, cooling requirements are met, and missile protection is achieved. No extensive review of shielding will be made. Shielding control has been simplified by a procedure which requires a hold in power at 10% of full power on startup until a survey can be made. If any new beam apparatus does not meet criteria developed by the experimenters and Health Physics, the reactor will be shut down and the apparatus removed or blocked to prevent the undesirable leakage.

For experiments involving the in-pile irradiation thimbles, a review of each type of experiment will initially be required by the Reactor and Critical Experiments Safety Committee. Prior to their review, the Research Coordination Group will have worked with the experimenters and designers so that they are able to make a recommendation concerning the experiment.

The Research Coordination Group Team which reviews HFBR experiments will include the Research Coordination Group Leader, the HFBR Operations Group Leader and Deputy Group Leader, the head of the HFBR Health Physics Group, and Reactor Division engineers familiar with the experimental facilities and reactor characteristics.

The relation between the Research Coordination Group Leader and the HFBR Operations Group Leader is a close one but necessarily a formal one with regard to information on experiments. The Research Coordinator must give written information to the Operations Group Leader regarding all experiments. He must also make reactivity and heating estimates in writing to the Operations Group Leader. The Operations Group Leader has the authority to delay operation until he is satisfied that the necessary information is available and that the installation of the experiment is such as to assure safe operation. By participating in the review of the experiment, both formal and informal information is given to the Health Physics Group Leader and the Operations Group Leader. The installation and removal of all experiments is monitored by the Health Physics Group.

13.1.5 HFBR MAINTENANCE. The maintenance of the HFBR is a cooperative and joint effort shared among the Reactor Operations Group, the Reactor Instrument Group, and the Reactor Mechanical Maintenance Group. The responsibilities of each group are clearly defined, and written procedures establish the assignments in detail.

The immediate authority to perform maintenance of any kind on the reactor, whether operating or shut down, and regardless of other approvals and assigned responsibilities, must be given by the Reactor Shift Supervisor in charge in order for him to properly discharge his responsibilities.

13.1.5.1 Direct Maintenance. The HFBR Operations Group Leader is responsible for all direct maintenance and repairs to the HFBR. Direct maintenance, for this purpose, is defined as repairs or correction to reactor components found not to be in proper working order. For providing direct maintenance, both the Instrument and Mechanical Maintenance Group Leaders are responsible to the Operations Group Leader.

13.1.5.2 Preventive Maintenance. Preventive maintenance responsibilities are divided among group leaders as follows:

- (1) The HFBR Operations Group Leader is assigned the responsibility for continually surveying process conditions in an effort to detect the deterioration of reactor components, or conditions which might lead to deterioration. Use is made of routinely compiled operating data and of special tests or checks.
- (2) The Instrument Group Leader is assigned the responsibility for testing, calibrating, and replacing components of the instrument and control systems of the HFBR as necessary to insure that these components perform according to design requirements. This is accomplished through a system internal to this group by which the work assignments are made to the Technicians, written procedures for standardizing the work are provided as necessary, technical data are provided, and the result of the work is recorded and reviewed by responsible Instrument Engineers.
- (3) The Mechanical Maintenance Group Leader is responsible for continually surveying the condition of major mechanical and electro-mechanical components. To accomplish this, he establishes the procedures for routine inspections of critical components, lubrication schedules, and special measurements. Use is made of the Tickler Card System to insure the routine testing and checking of these items.

13.1.5.3 Indirect Maintenance. The various Group Leaders do not have the authority to make changes or additions to the reactor facilities unless specific approval has been given by the Reactor Division Head. Proposals for changes or additions, normally prepared by the Group Leaders jointly, are submitted to the Division Head. Action by the appropriate safety committee, the Director's Office and, in some cases, the Atomic Energy Commission, is then requested as may be required depending upon the nature of the change proposed. The responsibility for obtaining proper approvals rests with the Reactor Division Head.

13.1.6 OPERATING GROUP DURING INITIAL LOADING, CRITICALITY, AND STARTUP TESTS. The organization and responsibilities of the HFBR Operations Group will be the same for the initial loading and tests as for normal operation, as given above. The Operations Group Leader and Reactor Division Head will be advised by senior members of the HFBR Project Staff and the Experimental Reactor Physics Division of the Nuclear Engineering Department, who will be in more or less continuous residence at the HFBR during the startup period.

13.2 ADMINISTRATIVE PROCEDURES AND CONTROLS

13.2.1 OPERATING PROCEDURES. The HFBR will be operated according to written procedures. The primary source of these instructions is the HFBR Operating Procedures Manual. Supplementary instructions are given in a Temporary Operating Procedures Logbook and in a Shift Instruction Logbook. A description of each of these manuals, together with the procedures for preparation, review, and approval, is given in Section 12.6.5.

13.2.2 INITIAL LOADING AND STARTUP TEST PROCEDURES. The initial fuel loading, critical approach, and all startup period tests will be carried out according to written procedures. These are described in Section 12.6.5.12, together with the procedures for preparation, review, and approval.

13.2.3 OPERATING RECORDS. Operating records for the HFBR are kept in a Shift Operations Logbook, in Console and Process System Shift Logsheets, on chart records from various instrument recording devices, in a Canal Logbook, and in a Fuel Movements Logbook. Fuel Charging and Discharging Record Sheets and the Tickler File Card System are both written instructions for certain procedures and a record of the work done. Each of these books is described in Section 12.6.5.

13.2.4 MAINTENANCE PROCEDURES. Maintenance work is carried out according to written instructions contained in the Reactor Maintenance Manual. The Manual is prepared by the Reactor Division and approved by the Head, Reactor Division. Copies are kept on file with the AEC and the Reactor and Critical Experiments Safety Committee.

Procedures for controlling maintenance activities to the extent that the reactor is in a safe condition during the period of maintenance are contained in the HFBR Operating Procedures Manual. These procedures are detailed for maintenance which is done during both operation and shutdown of the reactor and include the testing of the device worked on. The Reactor Shift Supervisor is responsible for insuring that all maintenance activities are in compliance with established procedures. Upon completion of maintenance and testing, he has the authority to place reliance upon the component. This primarily refers to work on reactor control rods and components of the safety and alarm circuits of the reactor.

The maintenance control procedures have been developed to provide nuclear

safety during both preventive and corrective maintenance operations. However, it must be borne in mind that it may not always be possible to foresee every possible way in which corrective maintenance situations may occur. It is the responsibility of the Shift Supervisors and Operations Group Leader to exercise continued awareness and to prevent work in any situation not covered by approved procedures until those procedures can be provided.

13.2.5 REPORTS.

13.2.5.1 Reactor Division Monthly Report. The Reactor Division Monthly Report is submitted to the AEC and the Reactor and Critical Experiments Safety Committee. This report provides information on all reactors (including the HFBR) other than the Neutron Source Reactor and critical facilities. It is prepared during the month following the report period and release is subject to the normal delays incidental to publication, including patent clearance, editing, and distribution. The report covers the following topics:

- (1) Operation histogram, tabulation of significant operating parameters, and fuel inventories.
- (2) Tabulation and analysis of all reactor down time, including details and explanation of any unscheduled shutdowns.
- (3) Special operating problems.
- (4) Maintenance difficulties with reactor components.
- (5) All pertinent changes made to the reactor.
- (6) Status of reactivity and unusual gains or losses.
- (7) Status of fuel and fuel changes.
- (8) Status of all experiments, including experiments added and removed.
- (9) Tabulation of irradiations made.
- (10) Status of process conditions.

13.2.5.2 Special Reports. Special reports are prepared and submitted to the Reactor and Critical Experiments Safety Committee as follows.

- (1) Unusual Occurrence Reports: These reports are initiated by the Reactor Supervisors in accordance with the HFBR Operating Procedures Manual. As the title indicates, any unusual occurrence may be the subject of such a special report. The Reactor Division Head transmits copies of these reports to the Reactor and Critical Experiments Safety Committee Chairman without delay.
- (2) Prompt Critical Reports: A report is made in the event the reactor achieves prompt criticality. Reports are also made to report occurrences in which prompt criticality would have been achieved by one additional circumstance.
- (3) Unscheduled Shutdown Reports: The cause of each unscheduled shutdown of the HFBR must be explained. The Shift Supervisors initiate a report form for each of these. Those which deal with Long Island Lighting Company electric power failure and shutdowns due to

fail-safe action of circuits resulting from component failure are reported only in the Monthly Reports. All other causes for reactor shutdown are reported promptly by forwarding a copy of the report form from the Reactor Division Head to the Chairman, Reactor and Critical Experiments Safety Committee.

- (4) Other Reports: Reports, either written or oral, are submitted to the Reactor and Critical Experiments Safety Committee as the occasion requires to cover proposed changes in procedures.

13.2.5.3 Criticality Inspection Monthly Report. Each month a member of the Reactor Physics Experimental Group is assigned to inspect the uranium storage areas in the HFBR to ascertain whether these areas are free from nuclear excursion hazard. The report is made to the Reactor and Critical Experiments Safety Committee.

13.2.6 EMERGENCY PROCEDURES. A local HFBR Emergency Plan has been prepared and integrated with the overall BNL Site Emergency Plan. The procedures of the HFBR Emergency Plan will be tested by annual drills. The BNL Site Emergency Plan has been coordinated with off-site police and fire departments to assure the efficient use of manpower and equipment should the need arise.

13.3 SERVICE ORGANIZATIONS

13.3.1 REACTOR HEALTH PHYSICS GROUP. The Health Physics organization is concerned with all actions involving radiation and general safety. Its activities are a line responsibility of the Deputy Director and the Chairman of the Instrumentation and Health Physics Department. The Health Physics Group assigned to the BGRR will also cover the HFBR. This group will consist of one Health Physicist, three professional level staff members of the Health Physics Organization, and three experienced Technicians. Of this staff, the Health Physicist will have over-all responsibility for both reactors. One professional man and two technicians will be assigned full time responsibility for the HFBR. The remaining four people on the staff will be available and trained for emergency, shutdown, and non-routine health physics coverage at the HFBR.

The primary areas of responsibility of the HFBR Health Physics Group are listed below. Where appropriate, in each area of responsibility, recommendations are made to those who may be subject to any exposure to radiation. All exposure experience is promptly reported to the proper supervisors.

- (1) Radiation Control: Monitor entries into high radiation areas, supply and evaluate film badges for all personnel in the reactor building.
- (2) Contamination Control: Monitor the building for alpha, beta, gamma and tritium contamination. Possible internal exposure will be monitored by urinalysis techniques.
- (3) Airborne Contamination Control: Monitor both the reactor building and the stack discharge for airborne radioactivity. Instruments will be available that will detect fission product, radioactive gas, or tritium air contamination.
- (4) General Safety: Act as representative for the BNL Safety Office in all matters concerning general safety.

- (5) Training: Train all new employees, and periodically retrain all employees in radiation protection. Subjects covered will be the Emergency Plan, tritium control, radiation control, and local health physics rules.
- (6) Records: Keep appropriate records of all activities involving radiation and general safety.
- (7) Waste Disposal: Provide waste disposal services as outlined in the BNL Safety Manual.
- (8) Environmental Monitoring: Maintain the program of on-site and off-site area surveys and extend it to include tritium hazards, where necessary.
- (9) Emergency Coverage: Carry out the duties assigned in the Emergency Plan.

The authority of the Health Physics Organization is largely unwritten, but its recommendations are rarely ignored. Failure to follow a Health Physics recommendation on work within the Reactor Division would create an untenable position for the persons involved in view of the close working arrangements which exist. Health Physics personnel have the same authority as any other staff member with regard to nuclear safety, i.e., to push any one of a number of locally mounted SCRAM buttons whenever they feel the situation demands it. Police type authority is not employed since the objective has been to insure a close relation between the safety and operations personnel. The responsibility for personnel safety follows the same lines as all other operating responsibilities. The responsibility for the safety of all experimenters lies within the various scientific departments.

During the 4-12 shift, 12-8 shift, and on weekends, the presence of a Health Physicist at the reactor is not required although one is always available on site. Since the Operations, Instrument, and Mechanical Maintenance Staff are qualified for radiation survey work, they can perform this function in the absence of the Health Physics representative or until he can be called. During shutdown periods for charging and discharging of fuel, changes in in-pile experiments, and reactor maintenance, Health Physics coverage is provided. It is the responsibility of the senior Health Physics representative at the HFBR to determine the number of Health Physics personnel required and to schedule the coverage. It is the responsibility of the Operations, Instrument, and Mechanical Maintenance Group Leaders to keep the Health Physics Group Leader informed of all operating and maintenance plans on a day to day basis.

When persons outside the Reactor Division are involved in the maintenance of the reactors, they are required to work according to a written "Health Physics Instruction". The Laboratory Procedures with regard to the "H. P. I." are detailed in the BNL Safety Manual.

13.3.2 MAINTENANCE ASSISTANCE FROM OTHER ORGANIZATIONS. "Intra-Laboratory Requisitions" for time, labor, and materials to be provided by shops and groups outside the Reactor Division are initiated within the Operations, Instrument, or Maintenance Groups. Approval for such requisitions must be given by the Division Head or HFBR Operations Group Leader. Personnel from other departments work under the direction of the Reactor Division Group Leader whenever their work involves reactor components or ancillary equipment.

The Plant Maintenance Division of BNL provides riggers, plumbers,

carpenters, electricians, and equipment maintenance personnel for appropriate jobs at the HFBR. The Shops Division will provide heavy machine shop work where needed, as well as special materials fabrication.

13.3.3 TECHNICAL ASSISTANCE. Following the startup and initial operating phase, when the reactor is achieving full power routine operation, the HFBR Project Manager, the Project Engineer and several of the project staff will continue their association with the HFBR in a consulting capacity.

The Experimental Reactor Physics Division and the Theoretical Reactor Physics Division of the Nuclear Engineering Department are among the laboratory groups who provide consulting services for the Reactor Division on a continuing basis. The extensive participation of these divisions in the HFBR design ideally qualify them for this service. The Mechanical Engineering Division of the Nuclear Engineering Department provides assistance as needed on special engineering problems.

13.4 SAFETY ORGANIZATIONS AND COMMITTEES.

The Deputy Director receives periodic reports from several organizations and committees concerned with nuclear, radiation, and industrial safety of the HFBR. These groups are listed below.

13.4.1 THE HEALTH PHYSICS ORGANIZATION monitors, audits and files reports on all activities involving radiation and industrial safety. Its activities are a line responsibility between the Deputy Director and the Chairman of the Instrumentation and Health Physics Department.

The responsibilities of the HFBR Health Physics Group are detailed in Section 13.3.1.

13.4.2 THE AD HOC REACTOR INSPECTION COMMITTEE annually audits the operation of the HFBR. The committee is appointed by and reports to the Deputy Director. The committee inspection reports are evaluated by the Reactor and Critical Experiments Safety Committee and are transmitted to the AEC by the Deputy Director.

13.4.3 THE LABORATORY SAFETY COMMITTEE advises the Deputy Director on matters of Radiation and Industrial Safety. It also conducts safety evaluations of new facilities or operations involving significant hazards other than nuclear hazards. The committee is appointed by and reports to the Deputy Director.

13.4.4 THE REACTOR AND CRITICAL EXPERIMENTS SAFETY COMMITTEE evaluates nuclear safety matters associated with critical assemblies, reactors, and with the storage of fissionable materials. This committee is appointed by and reports directly to the Deputy Director. Since its function is largely within the area of nuclear safety a complete description of this committee's function with regard to the HFBR is given here.

The committee is advisory to the Deputy Director and to the Head of the Reactor Operations Division. It is advisory to the Deputy Director on matters of nuclear safety policy, on matters of establishment of original rules and procedures for reactors, including safety analysis reports, and for all subsequent changes in approved rules and procedures which may possibly affect safety.

The Reactor and Critical Experiments Safety Committee does not carry out

inspections. Inspection is provided through self-inspection within the operating units, by the Health Physics Divisions, and by Ad Hoc Committees appointed by the Deputy Director annually to make inspections of each reactor. The findings of each Ad Hoc Committee are reported to the Deputy Director and are discussed also with the Head, Reactor Operations Division, and with the Reactor and Critical Experiments Safety Committee.

The Reactor Division Head is responsible for determining that operations are carried out within previously approved working rules and procedures and within the policies of the Laboratory. Questions of safe interpretation of changes which might be considered as exceptions are referred to the Reactor and Critical Experiments Safety Committee for advice. If the committee's recommendation does not involve a policy change, it communicates directly with the Reactor Division Head. Normally, work proceeds in accordance with the committee's recommendation. In the rare event where the Reactor Division Head wishes to disagree, he presents his case to the Deputy Director for adjudication.

The membership of the Committee consists of persons qualified by experience and training to appraise the Laboratory's activities with respect to the special criticality hazards which arise in the handling of large amounts of fissionable material and in the operation of reactors. Although the membership of the committee changes from time to time as conditions dictate, the aim of such changes is to staff the committee with qualified scientists and engineers having such experience and maturity as will contribute to the attainment of the Committee's objectives and provide for continuity of purpose. The Head of the Reactor Division is the only member of the committee from the Reactor Division. The current roster of the Committee membership is always on file with the Brookhaven Office of the AEC.

The committee meets at least once each calendar quarter on call of the Chairman. Written minutes are kept of the proceedings of the committee. The committee Secretary is responsible to the Chairman for maintaining appropriate records. A member of the Brookhaven Office of the AEC attends the meetings of the committee.

The committee reviews all proposed alterations to the HFBR which, in the opinion of any member of the committee or the Head of the Reactor Division, can affect reactor safety. The committee issues guidelines as to what reactor alterations and what reactor experiments must be brought to its attention. Those items specifically requiring the review of the Reactor and Critical Experiments Safety Committee are as follows.

- (1) Any change in operating procedure for, or any physical change made in an existing reactor which affects its nuclear safety.
- (2) All proposals for use of more than 250 grams of fissionable material.
- (3) The proposed movement into any area (other than reactor, vaults, and critical assembly areas), or the accumulation therein, of more than 250 grams of fissionable material.
- (4) In-reactor experiments exceeding specified limits. These limits have been developed as guides to the Research Coordination Group Leader and provide for his approval, without committee action, of certain types of experiments with low potential hazard. In the case of the HFBR, all in-reactor experiments will initially be reviewed by the Reactor and Critical Experiments Safety Committee. Any future change in this control will depend upon experience and the nature of the

experiments proposed.

(5) Proposed changes to operating manuals for the Reactors.

The Committee is flexible in that items for its consideration can be submitted by the Deputy Director, the committee Chairman, the committee membership, or the various laboratory organizations. The committee relies primarily upon the unit heads to bring before the committee all matters which in their judgment require review. The Deputy Director has instructed the operating unit heads to provide information to the committee Chairman even when there is a doubt as to the necessity for committee review, in order that the decision regarding the need for committee consideration may be made jointly by the Chairman and the unit head.

The Reactor Division Head is responsible for keeping the membership of the committee generally informed of the status of the HFBR with regard to routine operations, and specifically informing them of experiences which may be indicative of the possible development of an unsafe condition. All experiences which actually result in accident, damage, or injury as a result of a nuclear event are also reported. This information is provided by routine scheduled reports and by special reports made directly and immediately to the committee Chairman (see Section 13.2.5).

13.5 HIRING AND QUALIFICATION OF REACTOR OPERATORS AND SUPERVISORS

The HFBR training and qualification program is administered by the Reactor Training and Procedures Coordinator. Training records, including written examinations, are retained in each Operator's and Supervisor's file, together with information pertaining to reactor experience and qualification status. The entire program, including the training files, has been reviewed by the Brookhaven Office of the AEC.

Classroom lectures have been used extensively in the training of the additional staff required for the High Flux Beam Reactor. Many of the older operators have been retrained in this manner. Normally, however, the turnover of this portion of the staff is so low as to make this method of training impractical as a continuous process.

13.5.1 REACTOR OPERATORS

13.5.1.1 Hiring Qualifications. The minimum educational prerequisite is high school graduation. Whenever possible, technical school or Junior College graduates are recruited. Prospective Operators are required to make a score of 60 percent or better on the Test of Mechanical Comprehension, Form BB, by Bennett and Fry. This requires that the prospective Operator score as high as 60% of the first year college students who took the same test. The prospective Operator is given a thorough physical examination by the Medical Department. Insofar as possible this examination eliminates candidates whose general health and physical condition are such as to cause possible operational errors. Epilepsy, insanity, diabetes, defective hearing or vision, or any mental or physical condition which might cause impaired judgment or motor coordination may constitute sufficient cause for eliminating a candidate from consideration.

13.5.1.2 Probation Period. Prospective Reactor Operators are hired on a probationary basis of six months. The long probationary period is required for Reactor Operators since they are first trained by the Health Physics Organization and only after completion of this phase of the training is it possible for the Reactor Supervisors to observe their work. Ample time for judging their

abilities is therefore provided within the probationary period. (Other laboratory employees are normally hired for a three month probationary period).

The initial training of each new Operator by the Health Physics Organization is for the purpose of thoroughly grounding and indoctrinating the trainee in the radiation safety aspects of his work as a Reactor Operator. At the end of his training, the Health Physics Organization conducts a practical examination of the trainee which assures that he is able to compute working times, survey for himself the normal radiation hazards that he is likely to encounter in his work as a Reactor Operator, and that he is familiar with protective clothing, instrumentation, emergency breathing apparatus and other safety equipment. This initial phase of training provides the trainee with confidence that the laboratory is taking adequate steps to assure his safety, and at the same time, the laboratory gains confidence that the trainee is competent to recognize hazardous conditions and take corrective 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 crew. A training check list is used as a guide and to assist the Supervisors in conduct of the training. Classroom methods of instruction may be employed during this time, depending upon the number of people involved. At the end of this period, the trainee is given a practical examination under the direction of the Training and Procedures Coordinator. A decision is then made as to whether sufficient progress has been made to predict that the trainee will be able to progress eventually to "A" Operator. If the prediction is negative, the trainee may be transferred or terminated without recourse to the grievance procedure. If this prediction can be made in the affirmative, the Operator is retained and continues training.

13.5.1.3 "B" Grade Operator. No later than twelve months after hiring, the Operator is examined for his fitness for promotion to "B" Operator. In preparation for the examination he continues his training with an operating crew and, with the aid of an itemized check list, he and his Supervisors make preparations for his "B" examination. He must demonstrate his ability to perform the duties of a "B" Operator to the satisfaction of his Shift Supervisor prior to taking the "B" Operator examination.

The examination is primarily practical in nature and is conducted under the direction of the Training and Procedures Coordinator. The Operator is required to demonstrate ability to perform the duties of a Reactor Operator according to the Operating Procedures Manual. If the Reactor Operations Group Leader determines that he has successfully passed this examination on the basis of personal observation and the recommendations of the examiner and Supervisors, he will be promoted. If not, he will continue his trainee status for an additional six months at the end of which he will be re-examined. If he fails to pass the examination for "B" Operator for the second time, he is terminated. (This has never occurred in practice.)

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 for the specific reactor involved under indirect (i.e., readily available but not necessarily observing) supervision. In addition to providing a check on the individual's qualification, the "B" Operator examination also provides another opportunity for screening out those individuals exhibiting low potential ability for future advancement.

13.5.1.4 "A" Grade Operator. The achievement of "A" Operator status reflects possession of a degree of background knowledge, experience, 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 required in the case of a "B" Operator.

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

When the operator has been classified as Operator "B" for a period up to 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 all phases of reactor technology. The subject matter for this examination is specified in advance, but the questions are changed for each examination. The status of "A" Operator must be renewed at two year intervals by the successful completion of a written examination.

13.5.1.5 Medical Examinations. Retention of Reactor Operator status in any grade is contingent upon the maintenance of minimum physical and general health standards as determined by the Medical Department at intervals of not greater than two years.

13.5.2 REACTOR SUPERVISORS

13.5.2.1 Hiring Qualifications. The minimum education prerequisite for hiring for the position of Reactor Shift Supervisor is a Bachelor's Degree in some phase of engineering. Promotion to the position of Reactor Shift Supervisor from positions in which experience may be carefully weighed as a substitute for educational background may also be considered. Prospective Reactor Supervisors must meet the same physical and general health requirements as prescribed for Reactor Operators.

13.5.2.2 Probation Period. The probation period for all salaried employees is three months. Maintenance of employment is contingent upon becoming qualified as a Reactor Shift Supervisor within a reasonable length of time.

13.5.2.3 Health Physics Training. Initial phases of the training of Reactor Supervisors, as in the case of Reactor Operators, is with the Health Physics Organization. The same subject areas are covered, but in considerably greater detail. The Reactor Supervisor is required to pass a practical examination similar to the one given to Reactor Operators. In addition, he must pass a written examination.

13.5.2.4 Reactor Training. Training of Reactor Supervisors is largely "on-the-job" training, conducted by qualified Reactor Supervisors and members of the Reactor Instrument Group under the direction of the Training and Procedures Coordinator. The academic level or equivalent experience of the prospective Supervisor is such that he may be relied upon, with a reasonable amount of assistance from others, to obtain the required technical information from the documents, reports and manuals, etc., which are readily available. There is no specified time required for training Reactor Supervisors. When the Training and Procedures Coordinator determines that the Supervisor-in-Training is prepared for a certain phase of his qualifying examination, it will be given.

13.5.2.5 Reactor Supervisor Qualification Examinations. These examinations are given in five phases, each covering subject matter required for qualifying the Reactor Supervisor, with a certain amount of additional coverage to

test the breadth of knowledge in the particular phase tested.

- (1) **Health Physics Examination:** This written examination may cover the following subject matter: general health physics philosophy and policies, area monitoring, waste disposal, biological effects of radiation, maximum permissible exposure policy and practices, personnel monitoring, bio-assay, and local exposure and contamination control.
- (2) **Reactor Instrumentation Examination:** This written examination covers all phases of reactor instrumentation characteristics.
- (3) **Reactor Technology Examination:** This written examination covers nuclear physics pertinent to the reactor field, reactor physics, reactivity effects, control rod effects, etc.
- (4) **Examination on Reactor Operations:** This written examination covers the following subject matter: determination of reactivity worth, reactivity coefficients, elementary heat transfer, burn-up calculations, annealing procedures, major experimental facilities, water treatment procedures, rules and limits, and emergency procedures.
- (5) **Oral Examination by Qualifying Board:** The Reactor Division Head appoints a committee of at least three persons experienced in reactor procedures, reactor operating and supervisory problems, and management procedures. The committee reviews the written examinations given and further examines the candidate orally, as may be necessary, to determine the fitness of the candidate to perform the duties of a Supervisor. This examination may, in part, consist of the construction of hypothetical conditions to determine the reaction of the Supervisor in non-routine situations.

13.5.2.6 Re-Qualification. At intervals of two years each Reactor Supervisor is re-examined in writing as a condition of maintenance of qualified status. In addition, a medical examination is given at two year intervals to determine that acceptable physical and general health conditions have been maintained.

13.5.2.7 Advanced Training. All Supervisors are encouraged to improve their professional competence by advanced individual study. Selected individuals who demonstrate technical ability beyond the scope of the supervisory position they hold are, from time to time, given assignments of an engineering nature to broaden their experience and to provide a source of capable candidates for future openings in higher supervisory positions.

13.5.3 SPECIFIC HFBR TRAINING. The hiring of the additional Operators and Supervisors required for the HFBR, their training and qualification on the BGRR, and the final selection of the crew to be specifically trained for the HFBR has been completed six months in advance of the scheduled occupancy of the HFBR by the BNL staff. Fifteen qualified Reactor Operators and nine qualified Reactor Supervisors are receiving specific HFBR training. Seven of these Operators are of "A" grade, with many years of experience in operation of the BGRR. Five of the seven have also been qualified to operate the BMRR. The remainder of the Operators are fully qualified BGRR Operators with less experience. The average experience of the Reactor Supervisors is five years.

The specific training is divided into four parts, with final completion of the training scheduled just prior to the preoperational testing of the reactor.

The training program subdivisions are:

- (1) Formal training lectures on all phases of HFBR design, construction and operation. There are 24 persons assigned to present the lectures and each lecture is given twice. Many of the persons assigned are those involved in the design. Others assigned are from the supervisory staff. Four hours of lectures per week, over a period of four months, are required to cover the material. Examinations given at the conclusion of the lecture series become part of the individual's training file to support his final qualification as an Operator for the HFBR.
- (2) On-Site training on completed systems (while construction is in progress and concurrent with lecture series). Instruction in the fuel handling and control rod mock-up will be given during this period.
- (3) Light water flushing and cleaning of the plant, followed by light water testing. This phase will be extended to satisfy the training requirement that all operators are qualified to operate all process equipment under simulated nuclear conditions.
- (4) Heavy water operation of all process equipment, following the loading of heavy water. The adequacy of all operating procedures will be reviewed prior to loading fuel.

SECTION 14. ACCIDENT ANALYSIS

14.1 INTRODUCTION

The design of the HFBR has been guided as much by safety considerations as by the needs of the experiments. The primary goal has been a research reactor that can endure abnormal environmental conditions, system malfunctions, and human error without hazard to the neighboring populace or to the staff of the laboratory. The following sections discuss the sources of possible reactor accidents. The possibilities and consequences of each are analyzed to the extent necessary to determine the degree of potential hazard. It is shown that the design is conservative in the light of environmental hazards. Where equipment failure or human error could cause risks to the reactor or the public, redundancy of systems and successive safeguards have been introduced.

The self-regulating characteristics of the reactor are similar to those of other enriched uranium reactors. The temperature and void coefficients are everywhere negative and are essentially equal to those in other research and test reactors. In addition, the HFBR has a very large negative reflector temperature coefficient which, although delayed by about 4 seconds, would serve to temper all but the most rapid transients. HFBR also has the long neutron lifetime characteristic of heavy water moderated reactors. Thus, it would respond more slowly to a given reactivity insertion than would a light water reactor. It would be relatively difficult for a reactivity transient to cause violent physical damage to the HFBR since the time scales of physically credible transients are so long. On the other hand, transients would progress so slowly that the point of incipient burnout may be reached in transients in which SPERT light water experiments would show no burnout. The conservative steady state view on burnout discussed in Section 4.7.7 is, therefore, adopted in the accident analysis. It is concluded that all reactivity transients would be shut down by the safety system with no fuel damage (Section 14.2).

The most severe results of equipment failure or malfunction would be limited fuel plate melting, caused by channel blockage or by faulty fuel elements. Most of the resultant fission product activity in the primary coolant would be removed by the demineralizer system. Noble gases and some of the halogens would be released from the stack through the cover gas exhaust system. The possible extent of such release is discussed in Sections 14.3.1 and 14.3.2.

Shutdown coolant flow for heat removal can be supplied by any one of the two primary pumps or two shutdown pumps, with the two main coolers or the shutdown cooler. The shutdown pumps are driven from the normal electrical supply or from either of two emergency power systems. In the event of a complete failure of motive power, natural circulation paths open automatically in the vessel to allow shutdown cooling by natural circulation.

The relative vulnerability of spent fuel elements to overheating between the time they are removed from the vessel and the time they have been dropped into the canal leads to the possibility that equipment failure or human error could lead to fuel melting in this interval. Although the equipment has been carefully designed and the transfer mechanism is considered to be reliable, a fuel handling accident must be regarded as possible. The consequences are discussed in Section 14.5.

The system has been designed so that even a gross rupture of the primary

vessel or of the primary coolant lines will not uncover the core and cause fuel to melt. Nevertheless, it has been considered necessary to examine the consequences of an accident leading to complete loss of the primary coolant into the reactor building, with subsequent melting of the core and release of the fission products to the building atmosphere. This accident could only occur if there were extensive failures beyond a vessel rupture. Although the coolant temperature is well below the boiling point, slow evaporation and fission product afterheat would lead to a pressure buildup in the building. The pressure and temperature buildup and the resulting fission product leakage in this case have been calculated on conservative grounds. Under the worst expected meteorological conditions the calculated doses in the hospital and housing areas within the Brookhaven National Laboratory site, and in all areas outside the site are below the guidelines of 10 CFR 100.

14.2 REACTIVITY ACCIDENTS

The transient behavior produced by a sudden increase in reactivity has been analyzed by the methods discussed in Section 4.6.11. The analysis follows the time history of the reactor following a reactivity increase, taking into account the neutron behavior, the thermal behavior, and the feedback from the thermal processes to the kinetic equations. Section 4.6.6 discusses the safety rod system and the shutdown margin provided by this system. Automatic insertion of any one of the eight independent main rods will shut the reactor down once a transient has begun. Any four main rods will maintain the shutdown of the fully loaded core thereafter, even if the other four main rods and all of the auxiliary rods are fully withdrawn. The redundancy of safety rods and the large safety margins, coupled with the redundancy and conservative design of the electronic safety system lead to the view that all transients would be ended by safety rod insertion.

In the transient calculations it is assumed as a conservative view that only four of the eight main rods respond to a scram signal. The scram is assumed to take place as a result of an overpower trip at 50 MW, an instrument delay of 6 msec is taken, and safety rod insertion is assumed to follow the measured curve of position versus time (Figure 9.31). It should be noted that the calculated transients do not depend strongly on the number of safety rods that are assumed to act. Transients shut down by one rod would be only slightly more severe than those shut down by all eight. An additional feature of conservatism in the analysis is the assumption in most cases that safety rod insertion begins from the fully withdrawn position, where the differential reactivity worth is least. Further, the actual reactor instrumentation gives a scram trip on the combination of period and power level signals, rather than on power alone, and the shutdown of actual reactor transients would be more rapid than that assumed in the transient calculations.

14.2.1 CONTROL ROD WITHDRAWAL ACCIDENTS. The fastest rate at which reactivity is added to the reactor from the control rod system is from the ganged motion of 8 control rods at the critical position. At that time, the reactivity addition rate will be 0.083 %k/sec, or 10.6 cents/sec for the normal core and 0.097 %k/sec, or 12.4 cents/sec for a fresh-fuel core (see Figure 4.24). As fuel burnup proceeds, the control rod blades will be farther from the core, and the differential worth will accordingly be less.

The control rod withdrawal accident has been calculated on the assumption of reactivity addition at the constant rate of 12.5 cents/sec. Two initial conditions have been assumed; the reactor critical at low power (10 KW), and at full power (40 MW), at the start of steady control rod withdrawal. The calculated transient power histories for these cases are shown in Figures 14.1 and

and 14.2. The shutdown in both cases is from the scram action of 4 of the 8 main rods from the fully withdrawn position, with the overpower trip at 50 MW. In both cases the power rises a small amount above the trip point of 50 MW because of the delay in response of the trip circuits and the finite time for safety rod insertion. The overshoot is greatest for the transient starting from 10 KW, where the highest power reached is 58 MW. In this transient the burnout ratio has minimum values of 1.57 for the equilibrium core and 1.39 for the fresh-fuel core, both for the power distributions at the beginning of the respective cycles. No fuel damage would result in either case. The rod withdrawal transient from full power, 40 MW, peaks at about 51 MW, with correspondingly greater burnout ratios at the power peak (1.78 and 1.58) and no fuel damage.

14.2.2 IRRADIATION THIMBLE FAILURE ACCIDENT. Of the various experimental facilities and structural members in the reactor vessel, the irradiation tubes in the core center have the greatest reactivity effect. The table in Section 4.6.7 gives a measured reactivity worth of 85 cents (0.663 %k) for the filling of an empty central irradiation thimble with heavy water, as against 50 cents for flooding a beam tube. In the normal operation of the irradiation facilities the thimbles are all filled with heavy water from the experimental D₂O cooling system. There may be experiments in the future, however, in which helium cooling will be used in a voided thimble. To cover this possible experiment case, and to provide a transient calculation which is sufficiently severe to cover the flooding of the facilities which are always voided (the beam tubes), we assume that a central irradiation thimble has been operated with no heavy water and that an accident occurs which floods the thimble.

The reactivity input to the system in this accident has a total magnitude of 85 cents, and will be injected at a rate dependent on the size of the leak. An upper limit to the transient is set by assuming the 85 cent reactivity increment to be inserted instantaneously. An initial reactor power of 40 MW is also assumed, since the high initial power leads to more severe consequences in this case. The resulting power transient is shown in Figure 14.3. The power reaches a maximum value of 86 MW at 130 msec after the step reactivity input. The shutdown is due to the scram action of 4 of the 8 main rods from the fully withdrawn position, with an overpower trip at 50 MW. The minimum burnout ratio in the transient is 1.06 for the equilibrium core and no fuel damage would result. For the fresh-fuel core the burnout ratio at the power peak in the transient is 0.94 and the possibility of burnout at the hot spot exists. The time during which burnout might occur is short, however, only 65 msec, and the fresh-core loading is itself a relatively rare occurrence. In view of the conservative way in which both the transient and the burnout ratio are computed, we do not believe that fuel damage would occur in an actual thimble flooding accident.

The most likely beginning of such a transient, i.e., a small leak into the voided thimble, would produce a relatively slow reactivity addition rate and would lead to a scram if the automatic power control system could not counteract the reactivity addition rate fast enough. The scram action would result from process instruments or a high power trip, with essentially no overshoot because of the slow ramp rate of reactivity addition. Flooding of other facilities, such as a beam tube, an outer irradiation thimble, or an isotope thimble, would lead to the addition of smaller reactivities and less severe accidents.

14.2.3 BROKEN CONTROL ROD ACCIDENT. The main control blades are guided by glands in the transition plate. If a follower shaft should break, or the blade should by some other means become decoupled from the drive mechanism, the blade would fall to the fully inserted position alongside the core and introduce negative reactivity. While this failure would not be desirable, it would not cause a power increase. Rather, the reactor would shut down because of the reduction

of reactivity.

If the same kind of failure occurred to an auxiliary blade, the result would be an increase in reactivity as the blade fell away from the core. The rod calibration curves (see Figure 4.25) show that the maximum reactivity that would be added by complete removal of an auxiliary blade from the fully inserted position would be 0.87 %k, or 1.12 dollars. Transients have been calculated for the "broken" auxiliary rod accident with initial power levels of 10 KW and 40 MW. The auxiliary rod was assumed to move away from the core at twice the acceleration of gravity, starting from the fully inserted position. This motion results in full withdrawal of the auxiliary blade in 197 msec. The 2 g acceleration is a conservative upper limit to the acceleration which might be given to the auxiliary blade by the core pressure drop force if the follower shaft broke above the transition plate. A break in the follower or blade coupling below the transition plate would lead to an acceleration less than that of gravity.

The power transients which result from the broken auxiliary rod follower are shown in Figures 14.5 and 14.6. The shutdown in both cases was caused by the scram action of 4 of the 8 main rods from a position 7-1/8 in. above the top of the core, with an overpower trip at 50 MW. The power peak in the transient from 10 KW is 52 MW, at 3 seconds after the start of the reactivity input. The minimum burnout ratios are 1.75 for the normal core and 1.55 for the fresh-fuel core. No fuel damage would result. The transient from full power is more severe, with a peak power of 81 MW at 190 msec after the start of the transient. The minimum burnout ratio for the equilibrium core is 1.12, and no fuel damage would occur. The minimum burnout ratio for the fresh-fuel core is 0.99, and the possibility of burnout exists. The time interval at less than unit burnout ratio is only 15 msec, however, and we do not believe that fuel damage would occur in an actual broken rod transient.

14.2.4 LIGHT WATER CONTAMINATION. The reactivity variation with the fraction of H_2O in the D_2O has been discussed in Section 4.6.12. Increasing the fraction of H_2O slightly in the just critical reactor would lower the reactivity as the reflector is poisoned. Greater concentrations would lead to reactivity increases as the in-core neutron moderation is improved. If the D_2O were completely replaced by H_2O , the reactor with control rods at the critical position for D_2O moderation would be supercritical. Complete withdrawal of safety rods beyond these positions would only increase the reactivity slightly. Increasing the H_2O content of the reactor that is held subcritical by complete insertion of the safety rods would cause a monotonic increase in k_{eff} to a value of 1.07 at full H_2O .

Because the safety rods could not hold the H_2O moderated and reflected reactor subcritical, a careful study of the possibility of introducing light water into the system has been necessary. No light water lines exist in the reactor pit except for the thermal shield and shutter (part of the biological shield system) cooling lines. There are no light water connections to the primary system. There are no light water lines on the top of the reactor shield. The operations floor of the reactor has no sprinkler system for fire fighting. The elevation of the reactor top shield structure above the operations floor eliminates any possibility that massive flooding of the operations floor would overflow water into the reactor pit, even if flooding of this magnitude could occur. Most of the building would have to be filled first, and the hydrostatic head would burst the containment.

If the thermal shield coolant lines should leak or rupture, light water could enter the thermal shield cavity. No penetrations exist that would allow light water to enter the reactor vessel from the thermal shield cavity. The

appearance of water in the thermal shield cavity would be signaled by the presence of moisture in the CO_2 input and cavity drain line (see Section 7.7).

The pressurized region at the top of the reactor vessel is the only appreciable free volume in the primary system. This volume could accommodate only 190 gallons of light water. If, somehow, the entire 1350 gallon inventory of light water in the closed circuit thermal shield cooling system were to displace heavy water in the primary circuit, the D_2O would only contain 19% H_2O . The k_{eff} of the reactor with all rods inserted would be raised to a value of 0.90.

If the light water from the thermal shield cooling system were to dilute only heavy water in the vessel, the mixture would be 62% H_2O . This would be above the amount which the safety system could control. Higher concentrations still might result from dilution of selected regions of the vessel inventory. To reach such conditions, it would be necessary to postulate a failure in the thermal shield cooling lines which transferred water by a selective process to the vessel, with no mixing in the primary circuit, without detection through reactivity effects or the thermal shield process instrumentation, and in amounts greatly in excess of the free volume of the primary system. No such circumstances seem possible.

Similar remarks apply to the biological shield cooling system (and shutter cooling lines), except that in this case the system capacity is much lower (340 gallons). Even if the total volume of light water in the biological shield system were to dilute heavy water in the reactor vessel alone, the mixture would have only a 16% H_2O content, which is insufficient to overcome the shutdown margin of the control rods.

The places of closest contact of the primary system and any light water systems are the primary and the shutdown coolers. Although considerable care has been taken in fabricating and inspecting these components, the history of the field suggests that leaks are possible. During operation, when the primary system is pressurized, the substantial pressure differential between the primary and secondary sides of these components would force heavy water into the light water through any leaks, rather than the reverse. The activity detectors monitoring the secondary circuits would detect such leakage at very low levels (see Section 11.4.2), and corrective measures would be taken before a substantial problem developed. During shutdown, when the reactor is depressurized and the primary pumps are shut down, the pressure drops across the coolers are smaller but are still in such a direction as to cause any water leakage to be outward from the primary system. A reversal of pressure that could lead to leakage of light water into the heavy water system would occur if the primary loop were depressurized, with primary coolant pumps shut off, and secondary flow to the main coolers still running. This set of circumstances should never occur, and primary-secondary pressure difference instruments are provided for each cooler to detect the condition if it did exist. If the pressure reversal should occur, a small backleakage might result if there were leaks in the exchangers. Massive leaks at such times would not be possible because the pressures are low, and failures that would cause them would have been detected during prior operation through radioactivity in the secondary circuit. Furthermore, the addition of light water would be physically limited to the 190 gallon free volume of the primary system, and this amount would not be enough to affect the control rod worth and prevent reactor shutdown.

These considerations imply that substantial addition of H_2O to the primary system could not take place by any known means. Nevertheless, the large amount of excess reactivity that light water addition could produce has led to the view that additional measures to avoid the chance of a light water accident would be

prudent. Accordingly, the poison solution system discussed in Section 7.3 has been added. The cadmium concentration of this system is high enough so that mixing the contents of the tank with light water filling the entire primary system would provide a reactor of $k_{\text{eff}} = 0.9$ with all safety rods withdrawn.

14.2.5 COLD WATER ACCIDENT. The negative temperature coefficients of reactivity imply that if the core inlet water temperature were suddenly reduced, the HFBR would undergo an increase in reactivity. This increase would consist of a small rapid addition as the cold water passed through the core (50 ms passage time) and a larger increase over a period of about 4 seconds as the colder water coursed through the reflector region. A power transient would take place.

Two possible sources of cold water addition must be considered. The first would exist if there were secondary flow to the main coolers, cooling the heavy water contents of the main coolers, and the primary pumps were suddenly turned on. The colder heavy water from the primary coolers would be pumped into the vessel and core. Temperature, reactivity, and power transients would take place. However, these transients would only occur if the reactor were critical or near critical at the time the cold water reached the core. To avoid this set of conditions, interlocks have been provided to prohibit rod withdrawal unless primary water flow exists at normal levels, and also to prevent starting either main pump unless the rods are fully inserted (see Section 9.5). Thus, the reactor is not critical at times when the main coolant pumps are off. Surges of cold water from main coolant pump starts cannot cause a supercritical state of the reactor, because of the large shutdown margin provided by the safety system during the startup time.

During the initial loading and low power tests, the interlock preventing rod withdrawal without primary coolant flow will be bypassed. At this time, the fission product content of the fuel will be essentially zero, and no potential hazard to the public will exist from a cold water accident. While the interlock is bypassed, procedural controls will supplement the protection offered by the reactor safety system in safeguarding the reactor and the personnel working with it.

The second possible source of cold water injection would exist if the reactor were running at reduced power on one primary cooler and one primary pump, with the other half of the primary cooling circuit not being used. If the second loop pump were started, and the heavy water in the idle main exchanger were cold, a cold water accident could result. Again, however, interlocks on the starting of main pumps prevent such an accident from occurring.

To provide an upper limit to the transient resulting from a failure of the pump start interlock, a calculation has been made assuming an instantaneous startup of a cold loop. The cold loop stream is mixed with the operating loop stream and slug flow through the core is assumed. An initial power of 20 MW for single loop operation has been assumed (rather than 30 MW) to reduce the hot loop flow and increase the reactivity effect from starting the cold loop. The transient is shown in Figure 14.4. The shutdown is caused by the scram action of 4 of the 8 main rods from the fully withdrawn position, with an overpower trip level of 50 MW. The power transient is relatively slow, and there is essentially no overshoot past the trip setting at 50 MW. The minimum burnout ratios are 1.82 and 1.61 for the normal and fresh-fuel cores, respectively, both at the beginning of the cycle. No core damage would be produced by this transient, which is itself a conservative view of the consequences of a failure of the pump start interlock and of subsequent improper procedures in starting an idle loop.

14.2.6 PRESSURE EFFECTS DURING TRANSIENTS. Pressures have been measured in transients in the heavy water moderated SPERT-II. The results are given in the report IDO-16891. The neutron lifetimes of the SPERT-II assembly and of the HFBR are almost identical. The core temperature and void coefficients of reactivity of the HFBR are a little more negative than those of the SPERT assembly. Thus, the peak pressures associated with transients from low power in SPERT-II will be upper limits to the peak pressures from excursions of a similar nature in the HFBR.

Pressures were not measured in SPERT-II with reactivity inputs as low as the 85 cents that would be produced by fracture and flooding of a central irradiation tube of the HFBR. The peak pressure in the transient which most closely approached the HFBR case was 0.7 psi, produced by a reactivity input of 1.22 dollars. Extrapolation of the measured pressures implies that the HFBR transient would generate a peak pressure surge of about 0.25 psi for a step reactivity input of 85 cents at low power.

The SPERT transients have all been run starting from low reactor power. A transient initiated by a step reactivity input while the reactor is at high power would cause more severe pressure effects, because the initial prompt jump of power would cause changes in heating and expansion at a rate much more rapid than those produced by the asymptotic reactor period. An upper limit to the transient pressure from an excursion started by a step reactivity input at high power can be established as the peak pressure from that asymptotic period which is equal to the period of the prompt jump.

After a step input of 85 cents in the HFBR, the power will jump with an initial prompt period of about 100 msec. The peak pressure in SPERT-II transients at an asymptotic period of 100 msec was about 5 psi. The maximum pressure surge to be expected in the severest of the HFBR transients, then, is 5 psi. As noted in Section 4.2, the HFBR fuel elements are tested at pressure drops of about 62 psi as a routine acceptance test. The pressure-tube type of construction used in the element is very strong, and a substantially greater pressure drop would be required to produce serious deformation of the element. Extrapolation of the SPERT-II data indicates that pressure surges of a magnitude to produce major damage (~100 psi at the operating temperature) would require step reactivity inputs of about three dollars, with the step introduced at high reactor power. No structural component in the core vessel can produce reactivity effects more severe than the 85 cent step input mentioned previously. In an actual transient, the fast flow of D₂O through the HFBR core should serve to reduce transient pressure surges well below the upper limit of 5 psi derived above. The core pass time is only 50 msec, and the expanding fluid and steam would be continuously withdrawn from the core in times which are short compared to even prompt periods.

14.2.7 FUEL CHANGING CRITICALITY. Section 4.6.8 discussed critical experiments which have shown that no configuration of the fully or partly loaded core would have more excess reactivity than can be overcome by any five of the eight main control rods. Since loading and unloading will only take place when all sixteen main and auxiliary rods are fully inserted (except during the initial critical loading), criticality accidents in the vessel during fuel handling cannot take place.

Fuel handling in the canal will avoid criticality hazards by providing spent fuel storage in tested criticality-safe baskets which are stored in safe arrays (see Section 7.8.5). Elements are passed to the canal one at a time, and each element is placed in a tested criticality-safe rack before the next element is received. In the same manner, elements are moved one at a time from the

temporary storage rack to the cut-off saw, and from there to the final storage baskets. Once the cut elements are placed in the final storage baskets they are not removed until they reach the fuel processing plant.

14.3 EQUIPMENT FAILURES AND MALFUNCTIONS

A number of possible failures of equipment are discussed in this section. These are malfunctions or failures of equipment due to faulty manufacture or assembly, improper servicing, human error, loss of power, etc., which would produce abnormal operating conditions. These accidents do not involve reactivity transients, and have consequences less severe than the situations discussed in Sections 14.5 and 14.6.

14.3.1 FUEL CLADDING FAILURE. The methods of manufacture, inspection, and testing of the fuel elements are discussed in detail in Section 4.2. In spite of the care that is taken to assure uniformity of fuel alloy and cladding, adequacy of bond, etc., some statistical expectation of failure that could release fission products into the primary coolant will exist. This is particularly true of the possibility of pinhole leaks, which are the most common type of cladding failure. A second type of cladding failure which must be considered is the failure of the cladding-fuel alloy bond. Bond failures have generally occurred in the past as manufacturing defects which were undetected until the unbonded area burned out in reactor operation. The ultrasonic test of the full plate surface for all HFBR fuel plates has been incorporated in the manufacturing process to detect such conditions. The possibility exists, however, that an unbonded area might escape detection.

The fission product release to the primary coolant would be greater from burnout of an unbonded area than from a pinhole in the cladding. We shall, therefore, treat the burnout case in the balance of this section. Should burnout occur, the presence of fission products in the primary system above the normal trace amounts from element surface contamination would be signaled by the gas monitor, RRA-105, and subsequently by the water monitor, RRA-100 (see Section 9.6.3). The reactor would be shut down and the helium purge rate from the gas system reduced to a minimum, to reduce the venting rate of volatile fission products. The vent path for the volatile release leads through the particle filter, FD-105, the activated charcoal iodine filter, FD-105A, and finally through the building exhaust air filters. The nonvolatile fission products released to the primary coolant would be removed over a period of time by the heavy water cleanup system (see Section 7.5).

The only radiation doses to laboratory personnel and to persons off-site result from the release from the stack of the volatile fission products, notably the rare gases and the iodines. To establish an upper limit to the size and the biological consequences of the fission product release following an accident of this sort, the following analysis has been made. It has been assumed that burnout occurs over a 7-1/2 in. length of a hot fuel plate, which amounts to 2% of the fuel alloy in an element. The size chosen for the burned out area is quite arbitrary, but is many times larger than would be expected for an undetected spot of defective bonding. The fission product content of a hot plate is about twice that of the average in the element, so that about 4% of the inventory of the maximum burnup element is involved in the release. Table 14.3-1 lists the iodine and rare gas fission product inventories in the melted section of fuel plate.

Early experiments by Creek, Martin, and Parker (14.1) had shown that at the relatively low melting temperature of uranium-aluminum fuel alloy about 5% of the iodines and 15% of the rare gases are released in melting. However, current

investigations with high burnup fuel show much larger releases; 37.8% of the iodines and 97.9% of the rare gases (14.2). We shall assume in these calculations that 40% of the iodines and 100% of the rare gases are released upon melting of the fuel alloy. The quantities thus released in melting are listed in Table 14.3-1. In addition, we take no account of likely depletion of the iodine by chemical reaction with the metal surfaces in the primary system or with dissolved solids, or of adsorption on surfaces. It is assumed that all of the released iodine and noble gases appear in the helium cover gas in a time interval short compared with the decay times of the individual isotopes.

Assuming that no action is taken to stop the flow of cover gas from the surge space, the activity would be purged to the stack through the charcoal bed iodine filter FD-105A and the particulate filter FD-105. The characteristics of these components are detailed in Section 7.4, in Tables 7.4-2 and 7.4-3. The noble gases and the iodine surviving these filters would then be passed to the reactor stack through the silver-plated copper iodine filter IB-317 and the absolute filter IB-318 (see Section 3.4.3).

Oak Ridge tests of activated charcoal bed filters have shown an effectiveness for removing elemental iodine in excess of 99.9%. The filter IB-318 is designed to remove particulates greater than 0.3 micron in diameter with an effectiveness greater than 99.9%; the particulate filter FD-105 is somewhat rougher, having a 5 micron pore size.

Table 14.3-1 Volatile Fission Product Release in Fuel Cladding Failure Accident

Isotope	Inventory in Melted Plate Section (curies)	Release from Melted Alloy (curies)	Release from Stack (curies)
I-131	1724	690	34.4
I-132	3112	1244	62.4
I-133	4320	1728	86.4
I-134	5080	2032	102
I-135	4040	1616	80.8
Kr-83 ^m	306	306	306
Kr-85 ^m	956	956	956
Kr-87	1912	1912	1912
Kr-88	2504	2504	2504
Xe-131 ^m	15.2	15.2	15.2
Xe-133 ^m	106	106	106
Xe-133	4360	4360	4360
Xe-135 ^m	1224	1224	1224
Xe-135	476	476	476

Because of uncertainty about the form of released iodine, the overall removal capacity of the filter system for iodine is assumed to be 95% for the purpose of this analysis. The silverplated copper mesh filter IB-317 is assumed to be ineffective in removing iodine that has escaped the charcoal bed, since this

iodine component is unlikely to be in the elemental form.

According to the assumptions stated, all of the noble gases and 2% of the iodine contained in the burned out fuel plate section would be vented at the top of the reactor stack. Table 14.3-1 lists these quantities. The maximum thyroid and whole body gamma doses for the four meteorological conditions described in Section 2.3.2 are given in Table 14.3-2. The release in each case is from the reactor stack, and the atmospheric diffusion is described by the Sutton's functions shown in Figure 2.6. The conversions from cloud concentrations to thyroid and gamma doses are discussed in Section 14.6.2.3, and the values used for the various isotopes are given in the last two columns of Table 14.6-1 in that section.

Since the amount of fuel assumed to melt in the calculation is a gross overestimate of what might be expected of a fuel cladding failure, and the treatment of the features affecting release have been conservatively treated throughout, the actual consequences of a cladding failure would be much smaller than the relatively low exposures given in Table 14.3-2.

Table 14.3-2 Maximum Doses from Fuel Cladding Failure Accident

	Atmospheric Condition	Thyroid Dose	Gamma Dose	Distance of Max. Doses
		(rad)	(rad)	(meters)
I.	Weak neutral, inversion at stack top	0.35	0.021	2,600
II.	Weak neutral	0.18	0.011	2,600
III.	Strong inversion, typical night	0.013	0.00075	40,000
IV.	Normal lapse, typical day	0.10	0.0059	600

14.3.2 PLUGGED WATER CHANNELS IN FUEL ELEMENTS. Extensive fuel melting caused by foreign material blocking fuel element water channels is less likely than simple burnout. Incidents of this kind have occurred in other reactors, and although care will be taken to avoid their occurrence in HFBR, the possibility cannot be discounted completely. Fuel melting from the blockage of channels by large foreign objects would be expected only at the beginning of an operating period. At this time, a large object that had inadvertently been left in the vessel during fuel changing could become lodged at the top of the core, and when the reactor power reached a high enough value, fuel melting would begin to take place. A smaller object could conceivably block several channels of an element later in an operating cycle.

To assess the consequences of water channel blocking, we assume that a complete fuel element melts, and that the fission product inventory is that for a peak power element at the end of life. The melting and release conditions are assumed to be the same as in the fuel cladding failure calculation, Section 14.3.1. The volatile fission products which are vented will have inventories in the fuel alloy, in the release from the alloy, and in the release from the stack which are just 25 times those given in Table 14.3-1. The thyroid and

gamma doses are similarly increased by a factor of 25; they are given in Table 14.3-3.

Table 14.3-3 Maximum Doses from Blocked Water Channel Accident

	Atmospheric Condition	Thyroid Dose (rad)	Gamma Dose (rad)	Distance of Max. Dose (meters)
I.	Weak neutral, inversion at stack top	8.9	0.52	2,600
II.	Weak neutral	4.4	0.26	2,600
III.	Strong inversion, typical night	.32	0.019	40,000
IV.	Normal lapse, typical day	2.5	0.15	600

The blocked water channel accident has been calculated on the basis of a larger than expected fuel melt, a pessimistic view of the iodine filtration efficiency, and the assumption of no corrective actions to reduce the release. Even at the normal helium purge rate of about 0.01 cfm at system pressure and temperature, the fission product release would extend over many hours. The helium system volume is about 27 ft³; the helium passage time through the gas system at the above purge rate is 45 hours. The occurrence of the melting would be signaled promptly by the gas monitor, RRA-105, and a few minutes later by the water monitor, RRA-100 (see Section 9.6.3). After the reactor was shut down, the purge rate would be reduced, or even shut off if the stack exhaust air monitor indicated an excessive iodine release.

The gas purge is not immediately cut off in the event of a fission product release in the vessel because the bubbler-type liquid level instruments require some gas flow to operate properly. In an actual release from the fuel, we expect the double filtration of the effluent gas to reduce the iodine content in the stack discharge to about 1/50th of that given above (99.9% efficiency vs. 95%), with a corresponding reduction in the maximum thyroid doses. Should the stack gas iodine monitor indicate excessive iodine in an actual release case the following measures can be taken to control the release. The shut-off valve on the helium supply through the control valve PRCa-101B (see Section 7.4 and Figure 7.5) can be closed, and the pressure controller PRCa-101 set to 250 psig. These actions will close off the gas purge, and will allow some addition of gas to the helium volume from the liquid level instruments without a corresponding purge. The rate of flow in the level instrument bubbler legs can be shut off, and the level instruments operated on an intermittent basis. Further gas volume will be added to the system at a slow rate by the flow of heavy water to the storage tank through the water monitor RRA-100. With these measures being taken, the gas purge rate can be reduced to a very low level.

14.3.3 POWER FAILURE. Loss of electrical power from the normal commercial source may occur several times a year, usually for a duration of only a second or two. In such a power interruption the reactor is scrammed by the safety system (see Section 9.5). The main pump under-voltage circuits open the pump

breakers after 3 seconds at zero voltage. For most of the commercial power outages, the main pumps stay on the line and return to full capacity when the fault clears. In these cases the reactor may either be shut down in the normal manner, with full flow continued for 10 minutes and shutdown flow thereafter, or the reactor may be restarted. In the few seconds of reduced flow which occur the fuel temperatures rise a maximum of 11°F (see Section 4.7.11), and then decrease.

In cases of power failure where the main supply is out for more than 3 seconds, the main pump breakers open and the primary flow coasts down to the shutdown level. The fuel temperatures have a maximum rise of 11°F, followed by a rapid decrease to the shutdown level. The shutdown flow is delivered by either of the two shutdown pumps, GA-102A or GA-102B (see Section 7.1). Electrical power to the shutdown pump is supplied from one of three sources. The commercial AC power is supplied to the shutdown pump through a solid-state rectifier unit (the pump motors are 15 hp, 250 VDC) for normal service. Upon failure of the commercial AC source, the HFBR emergency generator set is started. When the generator reaches rated voltage, the shutdown pump is powered by it, again through the rectifier unit. During the generator startup time the pumps are connected to the third source, the 250 volt fan house battery bank. All of the required switching is fully automatic. If the generator fails to start, the shutdown pump runs on the battery bank for the life of the battery. With the assorted emergency loads on the fan house battery, the expected life is 2-1/3 hours. Some extension of the battery life can be made by switching off some of the emergency lighting load.

In the case of a prolonged failure of the commercial electrical supply coupled with a failure of the emergency generator set, the forced flow shutdown cooling of the core ceases when the fan house battery is exhausted. The flow reversal valves above the core open automatically at this time and the natural circulation cooling mode is established (see Section 7.2.5).

In the case of a simultaneous failure of all pump power sources the reactor is shut down by the safety system and the core goes into flow reversal and natural circulation cooling as soon as the coastdown time has elapsed. Fuel temperatures in the natural circulation cooling mode are relatively low; a few degrees above the atmospheric boiling point. As noted in Section 7.2.5, makeup water must be supplied to the vessel after about 5.4 hours.

In all cases of power failure the core is adequately cooled and no melting will occur. The only release of radioactive material associated with any of the power failure cases is that which results from venting the pressurizing gas. The venting is done automatically when the power sources to the main and shutdown pumps all fail. The radioactive material released includes tritium-bearing heavy water vapor and traces of volatile fission products from fuel element surface contamination. This release is treated in Section 11, where it is shown that the maximum integrated doses for the short release are negligible, of the order of 1×10^{-5} rad or less.

14.3.4 PRIMARY SYSTEM RUPTURE. Considered here are the consequences of breaks in the primary system, either in the reactor vessel or in the piping and equipment. Since the primary system is a low temperature, low pressure (150°F, 275 psig), relatively unstressed piping system, such breaks are not likely. Further, since the primary coolant is well below the atmospheric boiling temperature, a system break does not lead to the steam pressure problems of a water cooled power reactor system.

Breaks in the system may be separated into two classes: those ruptures of

the reactor vessel which would cause the core to be uncovered if it were not the shield pit containment, and all other ruptures. In all cases the core remains covered and no fuel melts.

A failure of the reactor vessel at the level of the core or below results in the thermal shield cavity filling with heavy water. The gas annulus between the thermal and biological shields will be flooded, as will the shutter cases and the beam tubes on the inner side of the seal plugs. The amount of heavy water in the reactor vessel is sufficient to fill these shield spaces to the level of the thermal shield cover. The occurrence of a vessel rupture would be signaled by the reactor liquid level instruments, and for a large break, by the core pressure drop channels as well. The reactor is shut down automatically by the safety system, and as the liquid level falls, the flow reversal system is tripped. The vessel is then vented to atmospheric pressure, the siphon break valve opens, and all primary pumps are turned off (see Sections 7.2 and 9.5.3). The flow reversal valves open after the pumps are turned off, and the afterheat is dissipated in the natural circulation cooling mode (see Sections 4.7.13 and 7.2.5).

Ruptures at other points in the primary system cause the vessel liquid level to drop, and lead to the same sequence of shutdown and natural circulation cooling as above. Breaks in the upper part of the vessel, or in the main coolant pipes above the gas seals result in the leaked heavy water collecting in the shield pit. The circumstances are similar to those above, but with the important difference that the primary containment for heavy water around the core is the (in this case) undamaged spherical section of the vessel itself. Piping or equipment breaks in the shielded cells beneath the reactor result in the leaked heavy water draining to the FA-101 sump. The possibility of siphon action emptying the lower portion of the vessel is eliminated by the siphon break valve, HCe-101C, which opens in response to the low-low liquid level signal.

If a rupture of the primary system is in a location where it can be isolated by closing valves or by a water seal, the steam produced in the core can be condensed in the shutdown cooler, as in the power failure case. The condensate then supplies the makeup requirement for as long as is necessary. In such a case the only radioactive materials which are released are the traces of volatile fission products from fuel element surface contamination and a small amount of tritium-bearing heavy water vapor. The fission products give negligible integrated doses, as noted in Section 11. The amount of tritium released is that in the pressurizing gas atmosphere, 0.97 curies, plus some evaporation from heavy water drained to the FA-101 sump or collected in the shield pit.

If a rupture of the primary system is in such a location that it cannot be closed off, then the steam produced in the core will escape to the exhaust gas system and stack. Heavy water could be drained from the rest of the primary system for use as makeup water, but a light water makeup is preferable in this case to limit the tritium release. The light water would be supplied through the poison tank (see Section 7.3), with the full inventory of poison solution used as the first 350 gallons of makeup. Since the poison compound would not be removed by the boiling, additional light water could be added as needed without further poisoning (see Section 7.2.5).

The amount of tritium released, and the concentrations and doses which result are estimated in the following way. The heavy water inventory left in the core region in a primary system rupture would be at least the 188 ft³ volume in the vessel below the outlet nozzle invert, and might be as much as the entire 302 ft³ normal operating volume. The 302 ft³ volume (8.55×10^6 cm³) is assumed, together with the 25-year tritium concentration of 1.7×10^4 μ c/cm³ (no makeup or

loss; see Figure 5.7). The total tritium inventory to be considered is then 1.45×10^5 curies. The release from the stack of all of the tritium due to boiling of the heavy water is assumed. The release would occur over a period of several days, so that the wind direction and meteorology conditions should vary substantially during the release period. We assume, however, that the meteorological conditions are constant, and calculate the maximum tritium doses for the four meteorological cases described in Section 2.3.2. The results are given in Table 14.3-4.

Table 14.3-4 Maximum Tritium Doses in Primary System Rupture Accident

<u>Atmospheric Condition</u>	<u>Cloud Exposure (c-sec/m³)</u>	<u>Dose (rem)</u>	<u>Distance of Max. Dose (meters)</u>
I. Weak neutral, inversion lid at stack top	1.45	0.191	2,600
II. Weak neutral	0.725	0.0955	2,600
III. Strong inversion, typical night	0.052	0.00686	40,000
IV. Normal lapse, typical day	0.405	0.0535	600

The conversion from integrated cloud exposure to dose in rem in Table 14.3-4 is done by the method of Pinson and Langham (14.3). A "quality factor" (the old RBE) of 1.7 rem/rad is used for the 5.7 Kev beta rays from tritium. The conversion is $1 \text{ c-sec/m}^3 = 0.132 \text{ rem}$.

14.3.5 LOSS OF PRESSURE. The reactor system is pressurized by the helium cover gas to suppress boiling in the core. The helium system (see Section 7.4 and Figure 7.5) includes a number of connections to the exhaust gas manifold which are normally closed by valves. The malfunction of a valve operator, or the inadvertent manual opening of one of these valves would rapidly reduce the primary system pressure. Similarly, a pipe break in the helium system would dump the pressurizing gas. The consequences of such an accidental loss of pressure while the reactor is at full power are considered in this section.

The cover gas pressure is monitored by three instruments which give an alarm at high (220 psig) or low (190 psig) pressure, and cause a scram in two out of three coincidence at a lower pressure (180 psig). At 120 psig in the cover gas, the low-low pressure interlock instruments cause the main pump breakers to open (see Sections 9.4.1.3 and 9.5.6). The gas pressure control system, consisting of an inlet valve, an exhaust valve, and a controller, is a relatively low flow rate system (see Section 7.4). In the event of a helium pipe break or the accidental opening of one of the vent valves, the control system would be unable to supply gas at a rapid enough rate to maintain normal pressure. The effect of the control system has been ignored in the analysis of the loss of pressure accidents.

The accidents to be considered are the malfunctioning of the vent valve HCe-102, the accidental opening of a vessel safety valve (SV-101A or SV-101B), the opening of the manual 2-inch gate valve in line P118, and various pipe breaks. In all of these cases the release of radioactive material is confined to the tritium-bearing water vapor in the gas volume and the traces of volatile

fission products from fuel element surface contamination. The concentrations and doses from this release are given in Section 11; the doses are negligible. No melting of fuel occurs.

The opening of the air-operated, spring-loaded valve HCe-102 results in the surge volume pressure transient shown in Figure 14.7. The time to vent the surge volume to atmospheric pressure is 30 seconds (14.4). The reactor pressure instruments initiate a scram at an elapsed time of 1.5 seconds (1 second to reach the 180 psig scram pressure and 0.5 seconds pressure switch response time). At 4.6 seconds the pressure reaches the pump interlock trip level of 120 psig, the main pump breakers are opened, and the primary flow starts the coastdown (see Figure 5.10 for the coastdown curve). As the pressure decreases, the saturation temperature at the core hot spot also decreases. The surface temperature of the hot spots for the freshly-fueled core and for the equilibrium core are shown in Figure 14.7. The maximum hot spot temperatures, occurring at the beginning of the cycle, are given. The hot spot temperatures remain at the full power values for slightly more than 1.5 seconds after the pressure begins to decrease (14.5). The drop in hot spot temperature is very rapid once the control rods start the scram. The reactor power transient is calculated using the measured delays in the safety system and the rod scram motion and reactivity characteristics shown in Figures 4.25 and 9.31.

The closest approach of the hot spot temperature to saturation is about 1°F for the freshly-fueled core and 12°F for the equilibrium core. There should be little concern even if the hot spot temperatures exceeded saturation by as much as 20 to 40°F, since the only result would be local boiling for a very short time. The hot spot temperatures rise from the value shown at 3 seconds in Figure 14.7 to a maximum of about 235°F as the primary flow coasts down to the 800 gpm shutdown level. The temperature and flow conditions at extended times are normal for the shutdown cooling operation of the system.

The pressure and temperature transients shown in Figure 14.7 are characteristic of breaks in the smaller pipes in the helium system as well as the opening of HCe-102. Most of the helium system piping is in the process area on the operations floor. It is connected to the vessel through relatively long pipe runs, and the flow resistance in the piping to a break point in the process area makes the pressure decay time long. Pipe breaks closer to the vessel, as well as an accidental safety valve opening, are covered by the analysis of the effects of an instantaneous opening of the 2-inch gate valve in line P118. This valve is located just outside the shield top, beside the safety valves. The vent path through P118 is short, the line is large, and the pressure decay time is correspondingly short.

The pressure and temperature transients which result from opening the 2-inch valve in line P118 are shown in Figure 14.8. The time to vent the surge volume to atmospheric pressure is 4.2 seconds. The reactor pressure instruments initiate a scram at an elapsed time of 0.62 seconds (0.12 seconds to reach the 180 psig scram level and 0.5 seconds pressure switch response time). At about the same time (0.62 sec) the pressure reaches the 120 psig level and the low-low pressure interlock opens the main pump breakers. The hot spot temperatures for both the freshly-fueled and equilibrium cores momentarily exceed the saturation temperature. The maximum rise above saturation is 24°F for the fresh-fuel core and 8°F for the equilibrium core. The minimum burnout ratio in the transient is 1.92, for the fresh-fuel core at the beginning of a cycle, and occurs at 0.65 sec after the start of the accident. As in the HCe-102 case, the fuel temperatures rise eventually to normal shutdown levels from the low values shown at 3 seconds in Figure 14.8.

14.3.6 INSTRUMENT SYSTEM MALFUNCTIONS. Most of the instrument system malfunctions which can occur either shut the reactor down or are covered by the multiple, parallel channel arrangements which are used. Safety system failures such as relay coil burnouts cause a shutdown because these components must be energized to operate the reactor. The welding of scram and setback relay contacts is possible, but the multiple contact arrangements and the conservative ratings of the relays make the disabling of the system from this cause very unlikely. Pump interlock malfunctions or switchgear failures which might cut off the coolant driving force do not lead to serious consequences, as is shown in the section on power failure cases (Section 14.3.3). The accidental opening of the depressurizing valve HCe-102 has been considered in the previous section. The remaining instrument system malfunction which must be examined in detail is that of an accidental tripping of the flow reversal system components.

The flow reversal system protects the reactor against the consequences of a failure of all pump electrical power or a major rupture in the primary system. It is actuated by the opening of all four sets of breakers for the main and shutdown pumps, and alternately by the low-low liquid level instruments. The system shuts down all four main and shutdown pumps, depressurizes the primary system, opens the siphon break valve, and leads to an opening of the flow reversal valves as the primary flow coasts down (see Sections 7.2 and 9.5.3). The consequences of all of these actions occurring with the reactor at full power are examined in this section.

A trip signal to the flow reversal system opens the siphon break valve and the pump breakers at about the same instant. The opening of the main pump breakers starts a scram with only a few milliseconds delay (just the response time of the scram relays). The depressurizing valve, HCe-102, is opened after the pump breakers open. The flow reversal valves do not open until about 5.5 seconds have elapsed, at which time the primary flow has coasted down far enough to permit them to open.

The surge volume pressure transient is identical to that shown in Figure 14.7, but the reactor shutdown occurs at a relatively earlier time. The temperature transients are shown in Figure 14.9. The saturation temperature decreases slowly for the first few seconds because the decreasing core pressure drop from the flow coastdown tends to offset the gas pressure decay. The result is a greater margin between the hot spot temperatures and saturation than in the case of loss of pressure alone, even with the initial fuel temperature excursions from reduced flow (14.5). The closest approach to saturation is about 6°F for the freshly-fueled core and about 22°F for the equilibrium core. As noted in the previous section, hot spot temperatures well above saturation for the short times involved would not lead to fuel melting.

The fuel temperatures drop to a minimum of about 155°F at 2 to 3 seconds after the accident initiation. At this point the core power has been reduced to the afterheat level while the flow is still about 60% of normal. As the flow coastdown continues, the fuel temperatures begin to rise from the minimum. The abrupt change in fuel temperature at about 5.5 seconds (Figure 14.9) is due to the opening of the flow reversal valves, which bypasses coolant to the reflector. The fuel temperatures rise to the atmospheric boiling point in about 30 seconds. Flow reversal occurs soon after, and the natural circulation cooling is established.

The only release of radioactive material in the accidental functioning of the flow reversal system is that involved in venting the vessel gas volume. This release is treated in Section 11, where it is shown that the doses are negligible. No fuel damage occurs in the accident. The recovery from the accident

is only a matter of clearing the instrument fault which gave the trip, and starting a shutdown pump to set up normal shutdown cooling for the reactor.

14.4 FIRE AND ENVIRONMENTAL HAZARDS

14.4.1 FIRE. The reactor building and internal structures are constructed of concrete and steel for the most part and are not generally flammable. Limited area fires are possible in overloaded electrical circuits. Since the reactor safety system is arranged with a view to the possible failure of both normal and emergency electrical power sources, such fires will not affect the safe functioning of the reactor safety mechanisms. Flammable materials in the building are limited to small quantities of laboratory solvents and similar materials, which could not give rise to a general conflagration. The emergency generator set, which is fueled by propane gas, is located outside the containment area of the building itself so that a fire in that region will not spread to the rest of the building.

14.4.2 EARTHQUAKES. The likelihood of serious earth tremors in the BNL area is small, as noted in Section 2.5. The building and reactor structures are designed for horizontal accelerations of 0.1 g as compared to the maximum recorded accelerations in the area of about 0.01 g. Damage to the reactor or building from seismic activity is, therefore, not expected.

14.4.3 FLOODS. Long Island is not subject to general floods, except near the sea shore 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 to the south. The general elevation of the laboratory is 70 to 90 feet above sea level. The base level of the reactor building is 93 feet above sea level. Flooding from the sea is thus impossible unless the entire island is submerged. The terrain in the BNL area is mostly flat, with a few low hills and no large rivers, so that fresh water flooding is similarly impossible.

14.4.4 HURRICANES. 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 (stack, cooling towers) are designed to withstand these wind loads with an ample safety factor. The great improvement in hurricane detection and tracking in recent years by government weather services, as well as the general weather surveillance program maintained by the BNL Meteorological Group ensure notice of approaching storms. It is standard practice at Brookhaven to shut down all reactors when a hurricane approaches the local area.

14.4.5 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 either could contribute to a reduced safety condition 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 exposure 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 distance apart of the reactors is about 300 meters, it is apparent that the number of neutrons entering one core from the other is truly infinitesimal.

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 show that under no

conditions does the negative pressure at the stack base go to zero (14.6). The least negative pressure observed in these tests was .16 in. H_2O . The stack is of such large size (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 should be no flow from one reactor to the other. There are, in addition, check valves at the discharges of the two exhaust fans in the HFBR which would block the exhaust duct in a reverse flow situation.

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 some flow up the stack from both systems.

It should be noted that the tendency of the stack to produce a useful draft even on hot days is amplified by the heating of the air by the Graphite Reactor fans if they are operating, or by 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. Further, any wind which might be present at the stack top provides an aspirator effect.

14.5 FUEL HANDLING ACCIDENT. The fuel handling sequence involves lifting an element from the reactor 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 carry away the afterheat. After the removed element has fallen through the fuel chute into the canal, natural convection cooling will again remove the afterheat. During the transfer, however, artificial cooling must be supplied if the element is to be out of the water for longer than a few minutes. This cooling is provided by the forced flow of air through the element during the transfer from the reactor vessel to the chute (see Section 7.8). It is possible that the forced air cooling system might fail in conjunction with a malfunction of the transfer apparatus. If the fuel element could not be moved, nor air flow restored, melting would occur in about 10 minutes with a subsequent release of fission products to the atmosphere of the shield pit.

The fuel handling accident is assumed to take place in the course of changing fuel 12 hours after shutdown after a normal operating cycle. It is assumed that one of the maximum-burnup elements is in transit from the vessel to the fuel chute when simultaneous failures of the transfer equipment and the cooling system occur. The fission product afterheat causes the element to melt in about 10 minutes, with the melt falling on metal surfaces and solidifying, or falling into the reactor vessel or the chute and canal. During the molten phase, fission products are released.

The important dose-producing components which are released from the melting of aluminum fuel are the iodines and the rare gases. Inventories of these fission products in the maximum burnup fuel element at 12 hours after shutdown are given in Table 14.5-1. From the experiments of Creek, Martin, and Parker (14.1, 14.2) the release from the molten aluminum fuel is assumed to be 40% of the contained iodine and 100% of the rare gases. The resulting release of iodines and rare gases from the molten element is given in Table 14.5-1.

The gaseous fission products are carried in the exhaust air stream from the shield pit to the shielded cells on the equipment level, and from there to the building filter bank. The release and the sweeping of the gaseous radioactive material through the exhaust system to the filter bank will take place in a few

minutes, so that no further decay of the material need be accounted for. The rare gases pass through the filter bank without loss and are discharged from the stack. The amount of iodine which is retained in the filter bank depends upon the physical and chemical form in which the material arrives in the filter elements. Elemental iodine should be removed with an efficiency of about 95% in the silver-plated copper-mesh iodine filter. Iodine absorbed on dust particles of greater than 0.3 micron diameter should be removed with essentially 100% efficiency in the absolute filters. Unfortunately, the conditions of formation of various forms of iodine are not well understood and it is impossible to say with any certainty whether or not the iodine discharge from a given accident will be completely filterable or not. For the purposes of this accident calculation we assume that 70% of the released iodine arrives in the filter bank in a form in which it is removed by one or another of the several filter elements. Thus, 30% of the iodine passes up the stack and is released to the atmosphere. The quantities of the iodines and rare gases emitted from the stack top are given in Table 14.5-1.

The results of the radioactivity release from the fuel handling accident are given in Figures 14.10 and 14.11. The thyroid and whole body gamma doses from the radioactive cloud are given for each of the meteorological cases of interest (see Section 2.3.2). The maximum doses occur for the weak neutral diffusion case with an inversion cap at the stack top, Case I. They amount to 45 rad for the thyroid dose and 0.20 rad for the whole body gamma dose. It should be remembered that Case I is a relatively rare condition, occurring 2 to 3% of the time at most, and with the further limitation that in most cases the Graphite Reactor will be operating and the effective release height of the plume will be much higher than the physical top of the stack. Case II, the second most severe dose condition is also relatively rare, occurring 1 to 2% of the time, and also with the amendment that in most cases the Graphite Reactor operation will reduce the maximum dose. The more common inversion and lapse conditions of Cases III and IV give lower doses: 12.5 rad for the thyroid dose and 56 millirad for the whole body gamma dose, in Case IV. In the dose calculations the conversion factors from cloud concentration (c-sec/m³) to dose in rads are calculated from data in TID-14844 (14.7). Values of the thyroid and whole body gamma dose conversion factors for the various isotopes of interest are given in Table 14.6-1, in connection with the discussion of the maximum credible accident.

14.6 MAXIMUM CREDIBLE ACCIDENT

14.6.1 BASIS FOR CHOICE. We turn now to the maximum credible accident which can befall the HFBR, in terms of the quantity of radioactive material released from the system. The maximum release would occur in the following circumstances:

- (a) The accident happens at the end of an operating cycle, when the fission product inventory is at the maximum value.
- (b) A complete melting of all fuel elements occurs, to permit the maximum release of fission products from the fuel alloy.
- (c) The reactor vessel ruptures and the shield pit is breached to uncover the core and permit the meltdown, and to allow a maximum release of fission products to the building atmosphere.

For such an accident to occur in the HFBR all of the carefully engineered provisions for emergency cooling of the core and for keeping the core immersed in heavy water must fail simultaneously. The design and construction of the reactor vessel, the shield pit, the experimental facilities, the shutdown cooling system, the natural convection cooling system, and the associated safety system instrumentation are specifically aimed at preventing this ultimate catastrophe.

Table 14.5-1 Fission Product Inventory in Maximum Burn-up Element
and Release From Fuel Handling Accident

Isotope	Inventory at Shutdown (curies)	Inventory 12 Hours After Shutdown (curies)	Release From Melting (curies)	Release From Stack (curies)	Thyroid Dose For <u>Unit</u> Sutton's Function (Rad)	Whole Body Gamma Dose For <u>Unit</u> Sutton's Function (Rad)
I-131	43100	43100	17200	5160	2673000	521
I-132	77800	77800	31100	9330	174000	4824
I-133	108000	86400	34600	10400	1456000	1498
I-134	127000	30	12	4	35	1
I-135	101000	31300	12500	3750	163000	1500
Kr-83 ^m	7650	995	995	995	0	11
Kr-85 ^m	23900	3590	3590	3590	0	169
Kr-87	47800	80	80	80	0	12
Kr-88	62600	3130	3130	3130	0	1684
Xe-131 ^m	379	379	379	379	0	16
Xe-133 ^m	2640	2640	2640	2640	0	160
Xe-133	109000	109000	109000	109000	0	2300
Xe-135 ^m	30600	9180	9180	9180	0	1239
Xe-135	11900	83300	83300	83300	<u>0</u>	<u>5806</u>
					4466000	19740

The probability of such an accident is felt to be exceedingly small in view of the measures taken to prevent it. We accept the circumstances listed above as the maximum credible accident for the HFBR because we cannot declare them impossible, even though we do not believe they will ever happen.

14.6.2 DESCRIPTION OF THE ACCIDENT. The description of the accident given below is divided into three parts. In the first part a brief description of the accident is given without attempting to assign numerical values to the many accident parameters. In the second part estimates are given of the sequence of events in the accident with regard to the evacuation of personnel from the building. In the third part a mathematical model of the accident is developed for use in determining radiation doses to persons at some distance from the building, particularly in the region beyond the site boundary.

14.6.2.1 Summary Description of the Accident. The reactor is assumed to be operating at full power a few hours before a shutdown for fuel changing at the time of the accident. At this time we assume the following events occur. First, a massive failure develops in the reactor vessel below the level of the active core, allowing the heavy water coolant to pour into the thermal shield pit. As the liquid level falls in the vessel the reactor is shut down by control rod action responding to the level instruments at "low" trip. As the level falls further the system is depressurized and the main pumps are turned off in response to the "low-low" trip. Some of the shield pit seals at or below the core level fail, so that the heavy water leaks from the pit and leaves the core uncooled. These seals may include beam tube seal caps, the thermal shield cooling pipe seals, thermocouple well seals, and the thermal shield cavity drain valves.

The afterheat causes the active portion of the core to melt. Volatile fission products are released and diffuse to the building atmosphere through the open pit seals. The nuclear incident alarm system responds to the presence of the fission products and causes the building air inlet valves to close, the air outlet valve to move to the throttling position, the air conditioning unit condensate drain valves to close, the exhaust fans to shut down, and an alarm signal to be sent to the Security Office. Heat and vapor from the heavy water spill and afterheat from the core cause the pressure of the building atmosphere to rise. The building atmosphere, containing fission products, leaks from cracks in the containment shell and from the stack at rates which depend on the overpressure. Radiation doses to persons outside the building are caused by the direct gamma radiation from fission products in the building and from exposure to the radioactive clouds from the containment shell leakage and the venting to the stack.

14.6.2.2 Sequence of Events in the Building. The onset of the accident will be signaled first in the control room by the liquid level instruments which will sound the low level alarm, scram the reactor, and depressurize the system as the level falls past the three trip points in the vessel. The pressure instruments will alarm and call for a scram and shutdown of the main pumps on low pressure as the helium supply system fails to keep up with the falling liquid level. As soon as the heavy water begins to leak from the shield pit, the area radiation monitoring stations in the building will alarm, both locally and in the control room, in response to the energetic gamma rays from N-16 and Na-24 in the water. Other reactor safety system instruments, primary flow and core pressure drop, will alarm and call for scram after the "low-low" level trip depressurizes the system and shuts down the primary pumps.

The evidence of a major reactor accident will thus be rapid and convincing and the Operator at the control console will use the paging system to order all

personnel to leave the building. There is no action which can be taken by the operating staff at this point to reduce the consequences of the accident. The best interests of both the personnel and the reactor system will be served by a prompt and orderly evacuation of the building.

Although the leakage of heavy water from the vessel to the shield pit may be at a high rate for a massive vessel failure, the leakage from the broken pit seals will be at a slower rate. The amount of fluid to be discharged is the 2200 gallon content of the vessel plus about 1000 gallons more thrown into the vessel by the primary and shutdown pumps before the former are switched off and coast down and the latter run dry at the suction. The driving force for the pit leakage is only a few feet of static head. The leakage paths in the beam tubes are through collimators of limited aperture and those through the drains, thermocouple wells, and thermal shield cooling pipe case are similarly tortuous. A total pit leakage rate of some 400 gpm from 6 or 8 failed seals seems sufficiently pessimistic to cover the worst case.

The perimeter fuel elements in the core, which have the highest afterheat rate, will begin to melt about one minute after they are uncovered. Volatile fission products will diffuse out of the pit shortly thereafter. It thus appears that personnel will have about 10 minutes to leave the building before any substantial quantity of fission products appear in the building atmosphere. In the circumstances, this should be adequate time for complete evacuation of the building. Some radiation dosage is to be expected from the activity in the heavy water which may appear on the experimental level and in the basement. Dose rates close to the leakage water will be of the order of several tens of rad/hr, but the time of exposure should not be long.

14.6.2.3 Calculational Model of the Accident. We develop here the calculational model for the building gamma ray dose, the building overpressure and leakage conditions, and the cloud exposures. As each parameter of the accident enters the argument it will be introduced under the title "Assumption 1, 2, etc.", and will be assigned a numerical value or functional form.

Assumption 1. The accident occurs with the reactor at full power at the end of an operating cycle, with the inventory of fission products which are significant in the dose calculations given in the column "Core Inventory" in Table 14.6-1.

The time chosen for the accident assures consideration of the maximum fission product inventory which can accumulate in the reactor. The selection of the iodines and certain of the krypton and xenon isotopes for consideration in the dose calculations is based on our survey calculations, which showed that only these components contribute significantly to the dose, and has precedent in the work of Di Nunno, Anderson, Baker, and Waterfield (14.7) and in many other reactor safety analyses.

The inventory numbers given in Table 14.6-1 are obtained from the fission product production curves of Blomeke and Todd (14.8), taking into account the fuel element rotation scheme in the HFBR (14.9). At the time of the accident the outer 14 fuel elements will have been in the reactor for 38 days at power. In the first 19 days they will have generated about one third of the reactor power, and in the second 19 days about two thirds of the power. The inner 14 fuel elements will have been in the reactor for 19 days, generating about one third of the reactor power.

Assumption 2. At the instant of the accident 7800 gallons of heavy water at 127°F issue from breaks in the vessel and shield and are distributed

Table 14.6-1 Fission Product Inventory and Isotope Dependant Parameters for the Maximum Credible Accident

i	ISOTOPE	CORE INVENTORY	BUILDING INVENTORY	LEAKAGE INVENTORY	DECAY CONSTANT	GAMMA ENERGY	THYROID DOSE CONVERSION	GAMMA DOSE CONVERSION	FILTER FACTOR
		(Curies)	S_{io} (Curies)	C_{io} (Curies)	λ_i (sec^{-1})	$E_{i\gamma}$ (Mev)	k_{iT} (Rad/c-sec/ m^3)	$k_{i\gamma}$ (Rad/c-sec/ m^3)	f_i
1.	I-131	$.87 \times 10^6$	$.435 \times 10^6$	$.218 \times 10^6$	$.996 \times 10^{-6}$	.39	518.	.101	0.30
2.	I-132	1.46 "	.73 "	.365 "	80.2 "	1.99	18.7	.517	0.30
3.	I-133	2.10 "	1.05 "	.525 "	9.25 "	.555	140.	.144	0.30
4.	I-134	2.77 "	1.39 "	.693 "	220. "	1.27	8.8	.330	0.30
5.	I-135	1.89 "	.945 "	.473 "	28.9 "	1.54	43.4	.400	0.30
6.	Kr-83 ^m	.158 "	.158 "	.158 "	101. "	.0415	0	.0108	1.00
7.	Kr-85 ^m	.535 "	.535 "	.535 "	44.1 "	.181	0	.0471	1.00
8.	Kr-87	.894 "	.894 "	.894 "	148. "	.56	0	.146	1.00
9.	Kr-88	1.19 "	1.19 "	1.19 "	69.5 "	2.07	0	.538	1.00
10.	Xe-131 ^m	.00734 "	.00734 "	.00734 "	.668 "	.163	0	.0424	1.00
11.	Xe-133 ^m	.051 "	.051 "	.051 "	3.49 "	.233	0	.0606	1.00
12.	Xe-133	2.22 "	2.22 "	2.22 "	1.52 "	.081	0	.0211	1.00
13.	Xe-135 ^m	.596 "	.596 "	.596 "	740. "	.52	0	.135	1.00
14.	Xe-135	.319 "	.319 "	.319 "	21.1 "	.268	0	.0697	1.00
15.	Solids	70.5 "	.705 "	.705 "	$71. \times 10^{-6}$ for 2 hours, t ⁻²¹ thereafter	.7	0	.182	1.00

uniformly on the floor of the building to form a water surface of 23,800 ft² area.

The spilled heavy water contributes to the overpressure condition in the building and it is our intent here to establish an upper limit for this factor. Thus, we take the full liquid inventory of the primary, shutdown, and purification systems (see Table 5.2-1) for the spill rather than the more reasonable estimate of 3200 gallons given in Section 14.6.2.2. The assumption of an instantaneous spill and distribution over the floor is made to simplify the overpressure calculation. It leads to a greater peak overpressure at an earlier time than would actually be the case and is, therefore, conservative. The water surface area chosen is approximately the area of the experimental level and is an upper limit to the evaporation surface which the heavy water could form. The temperature given is simply the mean temperature of the primary coolant at full power. The energy content of the spill is significant. With reference to the 70°F ambient temperature of the building the energy in the spilled heavy water is 4.07×10^6 Btu, or about 4300 MW-sec.

Assumption 3. At the instant of the accident the core melts completely and releases 50% of the iodines, all of the rare gases, and 1% of the solid fission products to the building atmosphere. Of the liberated iodine inventory one half remains in the atmosphere, available for leakage, and the other half adsorbs on internal surfaces.

This is the fission product release from a meltdown accident proposed in TID-14844 (14.7), and now widely used in safety analyses. The experimental work of Parker, et al., shows the release to be expected from high burnup aluminum alloy fuel is about 40% of the iodines and 100% of the rare gases (14.2). The difference in iodine release fractions is not great, and we adopt the larger release fraction for the maximum credible accident.

The assumed deposition factor of 50% for the iodines on surfaces in the building is supported by the experiments of Megaw and May (14.10) and of Morris, Diffey, Nicholls, and Rumary (14.11) on the behavior of iodine released in full-scale containment buildings. Their results would suggest a much greater deposition factor than that assumed here. However, the experiments were done with elemental iodine releases and extrapolation of the observed deposition rates to iodine from a core meltdown is not justified at the present time. The factor assumed here (50%) is felt to be conservative enough to cover the possible differences.

The inventory of the significant dose-producing isotopes in the building is given in the third column of Table 14.6-1 and the inventory available for leakage from the building is given in the fourth column of the table. The decay constants and gamma ray energies for each isotope are given in the fifth and sixth columns of this table. The decay behavior of the solids group is taken from TID-14844 (14.7).

Assumption 4. Heat transfer to the concrete floor and to the air from the water layer is assumed to be zero. The spilled heavy water evaporates at a rate given by

$$w = 0.156 (p_{\text{sat}} - p)^{1.2}, \text{ lb/hr-ft}^2 \quad (1)$$

where p_{sat} is the saturation pressure at the temperature of the water and p is the water vapor pressure in the building atmosphere, both in psia (14.12).

Evaporation at the rate given by eq. (1) occurs from the entire surface of

the spill. The total rate of evaporation is thus

$$W = 1.0313 (p_{\text{sat}} - p)^{1.2}, \text{ lb/sec} \quad (2)$$

with the same definitions as for eq. (1). The neglect of heat transfer from the water to the air above and the concrete below simplifies the calculation. The net effect is conservative since it leaves the water temperature higher at all times than would be the case if the additional terms were included, and since the energy current into the atmosphere by evaporation is several times the convective heat fluxes.

Assumption 5. The full afterheat of the core is carried away by the building atmosphere.

The afterheat of the core is given in Section 4.7.9, taken from recent work by Shure (14.13). The assumption that the full afterheat is taken up by the atmosphere gives a maximum overpressure effect from this source. Actually, one would expect the afterheat to be taken up in the thermal shield cooling system, which would presumably remain in operation throughout the accident. Or, if the shield cooling system should fail, the afterheat should be absorbed as a small temperature rise in the great mass of the biological shield. The energy put into the building atmosphere by this assumption is fairly large, amounting to 2.7×10^6 Btu (or 2850 MW-sec) in the first 80 minutes after the initiation of the accident.

Other heat sources were considered in the survey calculations and were discarded as factors in the overpressure calculation. The electrical load in the building is substantial but is removed by the air conditioning system. If, as one expects, the electrical power and the air conditioning continue in normal operation during the accident there is essentially no overpressure developed. The air conditioning capacity is so large that even if several of the eight main units are down the energy put into the atmosphere by the heavy water spill and the afterheat is promptly removed. One must keep in mind, in this connection, that the overpressure is developed by heating the atmosphere rather than by the addition of large masses of vapor, as is the case in water-cooled power reactor accidents. The efficiency of the air conditioning units and their energy removal capacity rise sharply as the ambient air temperature increases. In order to treat the maximum possible overpressure case we neglect the combined effects of the electrical load and the air conditioning system.

The other possible heat source which has been discarded is that from the outside environment on a very hot summer day. Its effect is trivial since the building atmosphere rises very rapidly in temperature in our calculation and is above 100°F in less than 13 minutes. The addition of the heat transfer terms to and from the environment thus result in a cooling effect soon after the initiation of the accident. Neither the energy input from the environment during the first few minutes of the accident, nor the energy loss thereafter are large enough to have any substantial effect on the overpressure condition.

Assumption 6. Heat transfer from the building atmosphere to 123,000 ft² of concrete surface occurs with a constant film coefficient of 0.20 Btu/hr-ft²-°F and a constant concrete temperature of 70°F.

With the above assumption the total heat rate from the atmosphere to the concrete is

$$q = 24600 (T_{\text{air}} - 70), \text{ Btu/hr} \quad (3)$$

where T_{air} is the temperature of the building atmosphere in °F. The film

coefficient used here is lower by a factor of about 5 than the actual natural convection air film coefficient for temperature conditions typical of the accident (14.14). This reduction is made to insure that the surface resistance to heat transfer is much greater than the conduction resistance in the concrete. It results in a lower heat rate than would actually be the case, and is thus conservative. The advantage of the low coefficient is that it may be taken as constant (the natural convection coefficient depends on the cube root of the temperature difference), and it allows us to assume that the concrete temperature does not vary with depth into the slab. The concrete thermal properties are: conductivity 1.5 Btu/hr-ft-°F, specific heat .23 Btu/lb-°F, and diffusivity .042 ft²/hr (14.15). The typical slab half-depth for the heat absorbing surfaces is 1 ft, so that the ratio of the surface resistance to the conduction resistance is

$$R_s/R_c = k/Lh = 1.5/1 \times .2 = 7.5$$

The surface area is essentially that of the floors and ceilings in the building, not counting the floor area covered by the water spill. It includes two faces of the concrete columns, but does not include any of the vertical surfaces of the many dropped beams in the floor structures. The total volume of concrete in the building is very large, amounting to some 250,000 ft³. The heat capacity of this great mass is 8.6×10^6 Btu/°F, so that the energy release in the heavy water spill, and from the afterheat for the critical first hour of the accident, could be absorbed with a temperature rise of about .8°F. Because the heat capacity of the concrete is so large, and its temperature rise during the accident so small, we simplify the calculation by assuming a constant concrete temperature. No account is taken in the calculation of the several million pounds of steel in the building dome, the craneways, and the various shields, although these elements have large surface areas in contact with the building atmosphere.

Assumption 7. The leakage rates from the building and the stack are proportional to the square root of the total overpressure, with the reference fractional leakage rates being 6.0×10^{-5} sec⁻¹ and 5.8×10^{-7} sec⁻¹ for the stack vent and building leakage, respectively, at 2 psi overpressure.

As pointed out in Section 3.6.2, the building leakage rate should contain terms proportional to the overpressure, as well as terms proportional to the square root of the overpressure. The assumption here of the square root dependence insures that the calculated leak rate will always be greater than the actual leak rate for the building for the overpressure range below 2 psi. The fractional leak rate values are those given in Section 3.6.1.

Assumption 8. Immediately after the air inlet valves close, the outside pressure decreases 0.1 psi.

This "barometric" term is added to the accident conditions to account for two effects in the outside atmosphere which may contribute to the overpressure in the building. The first is a reduction in the barometric pressure and the second is the pressure decrease on the lee side of the building due to wind. The wind effect is small, since we are concerned with meteorological conditions of poor diffusion. The ground level wind velocities for several cases considered in Section 2.3.2 range from 1 to 4 m/sec, or 2.24 to 9 mph. Even at the stack top the highest wind velocity is only 7 m/sec or 15.7 mph for these cases. The outside pressure decrease from the 7 m/sec wind is about 0.0042 psi. The remainder of the barometric term then accounts for a possible decrease in barometric pressure during the accident.

Assumption 9. For the radioactive cloud doses, the conversion from integrated exposure in c-sec/m³ to rads is given by the factors k_{iT} and k_{iY} in Table 14.6-1 for the thyroid and whole body gamma dose, respectively.

The thyroid dose conversion factors, k_{iT} , for the iodine isotopes are obtained from Table III (p. 25) of TID-14844 (14.7), using a breathing rate of 3.5×10^{-4} m³/sec, or 21 liters/min. The breathing rate is a mean value for adults during the active portion of the day (14.16). The whole body gamma dose conversion factors, k_{iY} , are obtained from the equation

$$k_{iY} = 0.26 E_{iY}, \text{ rad/c-sec/m}^3 \quad (4)$$

where E_{iY} is the energy of the gamma rays in Mev emitted by the i th isotope (14.17). This conversion factor is based on a calculation of the energy absorbed by a unit mass of tissue immersed in a semi-infinite cloud which emits gamma rays of energy E_{iY} . Thus, the cloud gamma dose is overestimated at points close to the source where the cloud has not diffused laterally to any extent.

Assumption 10. The building exhaust filters remove 70% of the iodines and none of the rare gases or solids from the stack vent gas stream.

In an earlier section on the fuel handling accident we have noted the difficulty of knowing the form in which the iodine release will arrive in the filter bank, and thus the uncertainty in assigning a removal efficiency for iodine. In that case an efficiency of 70% was assumed for the combined silver-plated copper mesh and absolute filters. The 70% removal seems to be reasonably conservative, and is adopted here for the stack release gas.

Similar questions of filter efficiency arise in the case of the solid fission products. These must exist in the form of very small particles to be airborne at all, and if they are smaller than 0.3 microns in diameter the absolute filters will pass them. In view of our inability to predict the particle size distribution, we assume that the filters remove none of the solids from the stack release. The rare gases are, of course, not filterable.

Assumption 11. Diffusion conditions outside the building are constant during the accident, and diffusion of the radioactive clouds from the building and stack is described by the Sutton's functions given in Section 2.3.2.

14.6.3 OVERPRESSURE AND LEAK RATE CALCULATIONS. With all essential assumptions now in hand we first calculate the overpressure condition in the building and the resulting leakage rates. This is a numerical procedure in which an account must be kept of the air, water, and water vapor inventories and their pressures and temperatures in successive finite time intervals. The calculation starts at zero time with the building atmosphere at the normal conditions of 70°F and 50% relative humidity, with the heavy water spill on the floor at 127°F, and with an overpressure of 0.1 psi from the barometric term (Assumption 8). Table 14.6-2 lists the input inventories, temperatures, and pressures, and gives the subsequent history of the overpressure condition (14.18).

The fractional leak rate for an overpressure of P psi is given by (Assumption 7)

$$\mu = 6.0 \times 10^{-5} (P/2)^{1/2}, \text{ sec}^{-1} \quad (5)$$

Note that although leakage occurs from the building as well as the stack vent, the latter is much larger (6×10^{-5} vs. $5.8 \times 10^{-7} \text{ sec}^{-1}$), and is used alone for the overpressure calculation. The specific heat of air used in the calculation

Table 14.6-2 Thermodynamic History During the Maximum Credible Accident

Time, Seconds	0	100	250	500	750	1000	1250	1500	2000
* Air Temperature, °F	70.00	77.03	83.35	93.23	99.67	106.08	110.20	114.02	119.56
Air Partial Pressure, psi	14.5144	14.6870	14.8066	14.9636	15.0008	15.0384	14.9951	14.9443	14.7908
Air Inventory, pounds	125660	125491	125041	124110	122986	121900	120671	119462	117104
Water Temperature, °F	127.00	123.83	119.70	114.06	109.92	106.76	104.25	102.23	98.89
Water Saturation Pressure, psi	2.0503	1.8810	1.6785	1.4323	1.2719	1.1602	1.0775	1.0148	.9180
Water Inventory, pounds	71189	70971	70688	70303	70022	69808	69639	69503	69279
Water Vapor Partial Pressure, psi	.1816	.2249	.2806	.3582	.4142	.4574	.4899	.5155	.5556
Water Vapor Inventory, pounds	978	1195	1474	1848	2112	2306	2452	2563	2736
Overpressure, psi	.1000	.3159	.4912	.7258	.8190	.8998	.8890	.8638	.7504
* Air Leakage, pounds	169	450	931	1124	1086	1229	1209	2358	2155
Water Vapor Leakage, pounds	1.31	4.28	10.98	16.73	20.31	23.24	24.57	50.59	50.34
Water Evaporation, pounds	218	283	385	281	214	169	136	224	153
Water Temperature Decline, °F	3.17	4.13	5.64	4.14	3.16	2.51	2.02	3.34	2.29
Convection Heat Loss, Btu	0	7206	22806	39685	50686	61638	68675	150401	169330
Afterheat, Btu	150700	142000	237000	180000	180000	151500	151500	270000	251000
Net Energy Into Air, Btu	150700	135000	214000	140000	129000	89900	82800	120000	81700

* The values in this group pertain to the time shown in the column heading.

* The values in this group pertain to the time interval beginning with the column heading under which they appear.

Table 14.6-2 (Cont'd)

Time, Seconds	2500	3000	3500	3750	4000	4250	4500	4750	4850
* Air Temperature, °F	123.38	125.72	127.03	127.21	127.53	127.57	127.61	127.43	127.36
Air Partial Pressure, psi	14.6144	14.4317	14.2620	14.1874	14.1274	14.0714	14.0270	13.9896	13.9807
Air Inventory, pounds	114949	113058	111480	110863	110335	109889	109536	109277	109220
Water Temperature, °F	96.60	94.95	93.69	93.21	92.77	92.38	92.02	91.70	91.58
Water Saturation Pressure, psi	.8564	.8141	.7832	.7716	.7612	.7521	.7436	.7364	.7336
Water Inventory, pounds	69126	69016	68932	68900	68871	68845	68821	68800	68792
Water Vapor Partial Pressure, psi	.5804	.5956	.6058	.6093	.6127	.6156	.6185	.6212	.6224
Water Vapor Inventory, pounds	2839	2902	2945	2961	2976	2990	3004	3018	3024
Overpressure, psi	.5988	.4313	.2718	.2007	.1441	.0910	.0495	.0148	.0071
* Air Leakage, pounds	1891	1578	617	528	446	353	259	56.7	
Water Vapor Leakage, pounds	46.70	40.51	16.30	14.09	12.02	9.59	7.10	1.56	
Water Evaporation, pounds	110	84	32	29	26	24	21	8	
Water Temperature Decline, °F	1.65	1.26	.48	.44	.39	.36	.32	.12	
Convection Heat Loss, Btu	182381	190376	97425	97733	98280	98348	98417	39244	
Afterheat, Btu	232000	218000	104500	104500	99500	99500	95000	38000	
Net Energy Into Air, Btu	49600	27600	7075	6767	1220	1152	-3417	-1244	

* The values in this group pertain to the time shown in the column heading.

* The values in this group pertain to the time interval beginning with the column heading under which they appear.

is that for constant volume, 0.1725 Btu/lb-°F (14.19).

The results of the overpressure calculation are shown in Figure 14.12. The fractional stack leak rate as a function of time during the accident is shown in Figure 14.13. The fractional building leak rate curve will have the same form as that shown in Figure 14.13, but with the ordinate scale multiplied by the ratio of the reference rates, i.e., by a factor of

$$\frac{5.8 \times 10^{-7}}{6.0 \times 10^{-5}} = 0.00967$$

The total leak rate, the sum of the building and stack vent components, has the same shape as the curve of Figure 14.13, with the ordinate scale multiplied by the factor 1.00967.

14.6.4 CLOUD DOSE CALCULATIONS. The time-dependent nature of the leak rates encumbers the cloud dose calculations with complex numerical integrations for each isotope considered. The calculational burden is reduced in the following manner. We determine the quantity of a non-decaying gas which leaks from the building in the "actual" release. Then we determine an equivalent release time, t_e , such that a constant leak rate, equal in magnitude to the maximum rate in the "actual" case, gives the same quantity of gas leaked in the interval $0 \leq t \leq t_e$. This procedure reduces the numerical integration problem to the single integral necessary to determine t_e . In terms of its effect on the calculated doses the approximation is conservative since we determine the equivalent release without regard to the decay of radioactive gases during the accident period, and treat the release as being at a more rapid rate at earlier times than in the "actual" case. The degree to which the doses calculated from the equivalent release are conservative is slight, because the principal dose-inducing isotopes have half-lives which are long compared to the total accident duration. In terms of the dose calculations, then, the equivalent release is a very good approximation to the "actual" release.

We establish the equivalent release as follows. Let the quantity of a non-decaying gas in the building at time t be M (air, for instance, with M in pounds). In the "actual" release case M is governed by the equation

$$dM/dt = -\mu_s M - \mu_b M \quad (6)$$

where μ_s and μ_b are the time-dependent fractional leak rates for the stack and building shown in Figure 14.13. Put

$$\mu_t = \mu_s + \mu_b, \text{ sec}^{-1} \quad (7)$$

and integrate:

$$M(t) = M_0 \exp \left(- \int_0^t \mu_t dt' \right)$$

The stack and building leak rates (time-dependent) are just μ_s and μ_b times $M(t)$, respectively. The total leakage from both sources during the accident, I_1 , is then

$$I_1 = M_0 \int_0^\infty \mu_t \exp \left(- \int_0^t \mu_t dt' \right) dt \quad (8)$$

In the constant leak rate, or equivalent release case, the individual stack and building fractional leak rates are the maximum values observed in the

"actual" case, μ_{sm} and μ_{bm} , and are constant. Their values (Figure 14.13) are

$$\begin{aligned}\mu_{sm} &= 4.04 \times 10^{-5} \text{ sec}^{-1} \\ \mu_{bm} &= 3.91 \times 10^{-7} \text{ sec}^{-1}\end{aligned}\quad (9)$$

and the total maximum fractional leak rate is

$$\mu_{tm} = \mu_{sm} + \mu_{bm} = 4.08 \times 10^{-5} \text{ sec}^{-1} \quad (10)$$

Repeating the calculation of the quantity of gas released, by now with constant leak rates:

$$\begin{aligned}dM/dt &= -\mu_{sm}M - \mu_{bm}M \\ M(t) &= M_0 \exp(-\mu_{tm}t)\end{aligned}$$

and

$$I_2 = M_0 \left[1 - \exp(-\mu_{tm}t_e) \right] \quad (11)$$

Here I_2 is the total quantity of gas released in the equivalent case up to the cutoff time, t_e .

We require I_1 to equal I_2 , and determine t_e from this relation:

$$t_e = -\frac{1}{\mu_{tm}} \ln \left[1 - \int_0^\infty \mu_t \exp \left(- \int_0^t \mu_t dt' \right) dt \right] \quad (12)$$

We find the double integral in the argument of the logarithm to have the value 0.1285 by a numerical calculation based on the leak rate curve (the number of significant figures is larger than the overpressure calculation justifies: we carry them only to avoid the accumulation of rounding errors). We then find, from eq. (12)

$$t_e = 3372 \text{ sec} \quad (13)$$

as the total duration time for the equivalent, constant leak rate release.

The calculation of the cloud doses is now made from the equivalent release model in the conventional way. We have two leakage sources whose fractional rates (for the stack and building) are given by eq. (9). These fractional rates are constant in time at the values given in eq. (9) in the time interval $0 \leq t \leq t_e$, and are zero thereafter. Each leakage source gives rise to a radioactive cloud, the doses from which are to be calculated separately.

The leakage inventory of the i th isotope in the building at time t , $C_i(t)$, in curies, is determined from the equation

$$dC_i/dt = -\lambda_i C_i - \mu_{sm} C_i - \mu_{bm} C_i \quad (14)$$

where λ_i is the radioactive decay constant of the i th isotope. Thus, with eq. (10), the time-dependent inventory is

$$C_i(t) = C_{i0} \exp \left[-(\mu_{tm} + \lambda_i)t \right], \text{ curies} \quad (15)$$

where C_{i0} is the initial leakage inventory of the i th isotope, in curies. The

release rates for the stack and building sources for the i th isotope are

$$\begin{aligned} Q_{is}(t) &= f_i \mu_{sm} C_{io} \exp \left[-(\mu_{tm} + \lambda_i)t \right], \text{ c/sec} \\ Q_{ib}(t) &= \mu_{bm} C_{io} \exp \left[-(\mu_{tm} + \lambda_i)t \right], \text{ c/sec} \end{aligned} \quad (16)$$

The quantity f_i in the expression for the stack release rate, $Q_{is}(t)$, is the filter factor. The filter factor has values of 0.30 for the iodine isotopes, and 1.00 for the rare gases and solids.

The quantities in eq. (16) are the source terms in Sutton's equation for the cloud concentrations. The product of a leak rate Q_i and the appropriate Sutton's function, $S(x)$, as defined in Section 2.3.2 is the cloud concentration for the i th isotope at time t (sec) and distance x (meters), $\chi_i(x,t)$, in cm/m^3 :

$$\begin{aligned} \chi_{is}(x,t) &= Q_{is}(t) S_s(x), \text{ c/m}^3 \\ \chi_{ib}(x,t) &= Q_{ib}(t) S_b(x), \text{ c/m}^3 \end{aligned} \quad (17)$$

Here we put subscripts "s" and "b" on the Sutton's functions to distinguish between the stack and building cloud diffusion characteristics. Explicitly,

$$\begin{aligned} S_s(x) &= \frac{2}{\pi c_y c_z u x^{2-n}} \exp \left[-\frac{(108)^2}{c_z^2 x^{2-n}} \right], \frac{\text{sec}}{\text{m}^3} \\ S_b(x) &= \frac{2}{\pi c_y c_z u x^{2-n}}, \frac{\text{sec}}{\text{m}^3} \end{aligned} \quad (18)$$

where 108 m is the stack height and the diffusion parameters are given in Section 2.3.2 for the cases of interest. The functions $S_s(x)$ and $S_b(x)$ are shown in Figures 2.6 and 2.7 of that section.

The instantaneous cloud concentrations given in eq. (17) are now integrated over the time interval $0 \leq t \leq t_e$ to obtain the total cloud exposures in units of c-sec/m^3 for each isotope, multiplied by the dose conversion factors k_{iT} or k_{iY} to convert the exposures to units of rads for thyroid or whole body gamma doses, and are summed over all isotopes. Four dose equations result, of which the stack cloud thyroid dose equation is typical:

$$D_{sT}(x) = \sum_i k_{iT} \int_0^{t_e} Q_{is}(t) S_s(x) dt, \text{ rad} \quad (19)$$

The other three equations, for D_{bT} , D_{sY} , and D_{bY} , are similar, with appropriate subscript changes. We substitute eqs. (16) into the dose equations and perform the integration, which is trivial since the time-dependent part of the integrand is just $\exp \left[-(\mu_{tm} + \lambda_i)t \right]$, with μ_{tm} and λ_i both constant. The results are the four final dose equations given below. A summary of the symbol definitions is appended for convenient reference

$$D_{sT}(x) = S_s(x) \sum_i f_i k_{iT} \mu_{sm} C_{io} E_i / (\mu_{tm} + \lambda_i), \text{ rad} \quad (20)$$

$$D_{bT}(x) = S_b(x) \sum_i k_{iT} \mu_{bm} C_{io} E_i / (\mu_{tm} + \lambda_i), \text{ rad} \quad (21)$$

$$D_{sY}(x) = S_s(x) \sum_i f_i k_{iY} \mu_{sm} C_{io} E_i / (\mu_{tm} + \lambda_i), \text{ rad} \quad (22)$$

$$D_{bY}(x) = S_b(x) \sum_i k_{iY} \mu_{bm} C_{io} E_i / (\mu_{tm} + \lambda_i) \quad , \text{ rad} \quad (23)$$

$$E_i = 1 - \exp [-(\mu_{tm} + \lambda_i)t_e] \quad (24)$$

- $D_{sT}(x)$: Total integrated stack cloud thyroid dose, rad.
 $D_{bT}(x)$: Total integrated building cloud thyroid dose, rad.
 $D_{sY}(x)$: Total integrated stack cloud gamma dose, rad.
 $D_{bY}(x)$: Total integrated building cloud gamma dose, rad.
 x : Distance from source to dose point, meters.
 $S_s(x)$: Stack cloud Sutton's function, eq. (18), sec/m^3 .
 $S_b(x)$: Building cloud Sutton's function, eq. (18), sec/m^3 .
 f_i : Stack release filter factor, Table 14.6-1.
 k_{iT} : Thyroid dose conversion factor, Table 14.6-1, $\text{rad}/\text{c-sec}/\text{m}^3$.
 k_{iY} : Gamma dose conversion factor, Table 14.6-1, $\text{rad}/\text{c-sec}/\text{m}^3$.
 i : Index number for the radioisotopes.
 μ_{sm} : Fractional leak rate for stack, eq. (9), sec^{-1} .
 μ_{bm} : Fractional leak rate for building, eq. (9), sec^{-1} .
 μ_{tm} : Total fractional leak rate, eq. (10), sec^{-1} .
 C_{io} : Initial leakage inventory, i th isotope, Table 14.6-1, c.
 λ_i : Decay constant, i th isotope, Table 14.6-1, sec^{-1} .
 t_e : Release duration time, eqs. (12), (13), sec.

The numerical work on the portions of the dose equations involving the radioisotopes is shown in Table 14.6-3. The factors E_i , defined in eq. (24) and given in the table for each isotope are the ratios of the radioactivity which is released in the accident as we calculate it here to that which would be released if the maximum leak rates were continued for all time. Thus, the shorter the half-life of an isotope, the more nearly the 3372 sec release time resembles the infinite duration release, and the larger the factor E becomes. This is shown in the case of Xe-135^m, which has a half-life of about 15 minutes, and a corresponding E value of 0.928. For isotopes of long half-life the quantity E is reduced, approaching a lower limit at $\lambda = 0$ of 0.1285. It will be noted in Table 14.6-3 that the E values for the long-lived isotopes I-131 and Xe-131^m (half-lives of 8.05 and 12 days) do indeed fall close to the lower limit.

The total thyroid and gamma cloud doses are shown in Figures 14.14 and 14.15 for the four meteorological conditions discussed in Section 2.3.2. Each curve is the sum of the building and stack contributions. The building cloud doses contribute the straight-line sections (compare Figure 2.7 of Section 2.3.2) and the stack doses the peaked sections (compare Figure 2.6 of Section 2.3.2).

Finally, the dose due to inhalation of tritiated water vapor which leaks from the containment building and stack is considered. Table 14.6-2 shows that a total of about 350 pounds of water vapor leaks from the building during the accident. The vapor is a mixture of light and heavy water vapor, continually becoming richer in D₂O vapor as the spill evaporates. Knowing the leakage fraction and evaporation rate throughout the accident enables one to keep an

Table 14.6-3 MCA Doses with Unit Sutton's Function, $\text{rad-m}^3/\text{sec}$

		E_i	Building Dose		Stack Dose	
Isotope			Thyroid	Whole Body	Thyroid	Whole Body
i			$\frac{k_i \mu_{bm} C_{io}}{\mu_{tm} + \lambda_i} E_i$	$\frac{k_i \gamma_{bm} C_{io}}{\mu_{tm} + \lambda_i} E_i$	$\frac{f_i k_i \mu_{sm} C_{io}}{\mu_{tm} + \lambda_i} E_i$	$\frac{f_i k_i \gamma_{sm} C_{io}}{\mu_{tm} + \lambda_i} E_i$
		$[1 - e^{-(\mu_{tm} + \lambda_i)t_e}]$				
1.	I-131	.130	.1378 x 10 ⁶	26.87	4.299 x 10 ⁶	840
2.	I-132	.334	.007345 "	203.1	.2291 "	6336
3.	I-133	.154	.08855 "	91.10	2.763 "	2842
4.	I-134	.584	.005315 "	199.3	.1658 "	6216
5.	I-135	.208	.02394 "	220.6	.7470 "	6882
6.	Kr-83 ^m	.379	0	1.777	0	185
7.	Kr-85 ^m	.248	0	28.73	0	2988
8.	Kr-87	.470	0	126.5	0	13160
9.	Kr-88	.309	0	699.5	0	72710
10.	Xe-131 ^m	.129	0	.3798	0	40
11.	Xe-133 ^m	.138	0	3.775	0	393
12.	Xe-133	.132	0	57.30	0	5958
13.	Xe-135 ^m	.928	0	37.17	0	3865
14.	Xe-135	.187	0	26.27	0	2732
15.	Solids	.313	0	140.1	0	14560
			.2630 x 10 ⁶	1862	8.204 x 10 ⁶	139707

account of the amount of light and heavy water vapor leakages as functions of time. By a stepwise accounting procedure the total amount of D_2O vapor that leaks during the accident is found to be 222 pounds. This represents about 1550 curies of tritium, with 15 curies leaking from the building and 1535 curies leaking from the stack. The tritium concentration of .017 c/cm³ D_2O , occurring after 25 years and assuming no D_2O makeup, was used for the release.

The tritium doses are negligible at the distances of interest in the maximum credible accident. To establish the magnitude of the tritium doses we consider meteorological Case I, which gives the maximum effect of the stack release, and Case III, which gives the maximum building release doses. For Case I the tritium dose is evaluated at 2600 meters, which is the distance of the maximum ground concentration for the stack release. The total cloud exposure is 0.016 c-sec/m³, including both stack and building cloud concentrations. Using the standard conversion of 1 c-sec/m³ = 0.132 rem (14.3), the total integrated tritium dose is 2.1 mrem. For Case III the tritium dose is evaluated at 200 meters to see the effect of the inversion cloud from the building at short distances. The total cloud exposure is 0.11 c-sec/m³ for this case, and the total integrated tritium dose is 15 mrem.

14.6.5 DIRECT GAMMA DOSE CALCULATIONS. There remains the calculation of the direct gamma dose from fission products in the building. For this calculation we ignore the depletion of the building radioactivity load by leakage, but include the decay of the fission products. The initial quantities of the significant dose-producing isotopes in the building are given in Table 14.6-1, under the column "Building Inventory, S_{i0} ". The quantity of the i th isotope at time t (sec) after the initiation of the accident is then

$$S_{i0} \exp(-\lambda_i t), \text{ c}$$

and the energy emission rate is

$$3.7 \times 10^{10} E_{i\gamma} S_{i0} \exp(-\lambda_i t), \text{ Mev/sec}$$

(Recall that 1 curie = 3.7×10^{10} disintegrations/sec.)

We take the emission to be isotropic, and the only shielding between the source and the dose point to be air. Then at distance x meters, the energy flux is

$$\phi_i(x, t) = \frac{3.7 \times 10^{10} E_{i\gamma} S_{i0} B_i \exp[-(\lambda_i t + \beta_i x)]}{4\pi x^2}, \text{ Mev/m}^2\text{-sec} \quad (25)$$

where B_i is the buildup factor, β_i is the linear absorption coefficient for air at the energy of the gamma rays from the i th isotope (in m^{-1}), and λ_i and $E_{i\gamma}$ are the i th isotope decay constant and gamma energy defined in the cloud calculation.

The energy flux is converted to a dose rate by multiplying by the energy absorption coefficient for air, β_{ie} (in m^{-1}), dividing by the density of air (1293 gm/m^3), multiplying by the conversion from Mev to ergs ($1.6 \times 10^{-6} \text{ ergs/Mev}$), and dividing by the conversion from ergs/gm to rad (100 ergs/gm-rad). The result of this arithmetic is the instantaneous dose rate from the i th isotope:

$$D_i(x, t) = \frac{.4578 \beta_{ie} E_{i\gamma} S_{i0} B_i \exp[-(\lambda_i t + \beta_i x)]}{4\pi x^2}, \text{ rad/sec} \quad (26)$$

The buildup factor is a function of the gamma ray energy (through the absorption coefficients) and of the distance and includes the effects of "sky-shine". We use the linear form (14.20):

$$B_i = 1 + \alpha_i \beta_i x \quad (27)$$

where α_i is the linear absorption constant. It is given by

$$\alpha_i = (\beta_i - \beta_{ie}) / \beta_{ie} \quad (28)$$

The total integrated dose is now obtained by replacing B_i in eq. (26) with the quantity given by eq. (27), integrating eq. (26) over the exposure time interval $0 \leq t \leq \tau$, and summing over all isotopes. The result is

$$D(x, \tau) = .0364 \sum_i \beta_{ie} E_{i\gamma} S_{io} F_i(x) G_i(\tau) \quad , \text{ rad} \quad (29)$$

$$F_i(x) = (1/x^2) (1 + \alpha_i \beta_i x) \exp(-\beta_i x) \quad , \text{ m}^{-2} \quad (30)$$

$$G_i(\tau) = (1/\lambda_i) [1 - \exp(-\lambda_i \tau)] \quad , \text{ sec} \quad (31)$$

$D(x, \tau)$: Total integrated direct gamma dose, rad.

x : Distance from building to dose point, meters.

τ : Exposure time, sec.

β_{ie} : Energy absorption coefficient for air, Table 14.6-4, m^{-1} .

β_i : Linear absorption coefficient for air, Table 14.6-4, m^{-1} .

α_i : Linear absorption constant, eq. (28), Table 14.6-4.

λ_i : Decay constant, i th isotope, Table 14.6-1, sec^{-1} .

$E_{i\gamma}$: Gamma energy, i th isotope, Table 14.6-1, Mev.

S_{io} : Initial building inventory, i th isotope, Table 14.6-1, c.

The direct gamma dose continues until the fission products in the building all decay. We shall calculate the dose in the first 2 hours and the dose for infinite time (except for the solid fission products). For the solid fission products the decay behavior is assumed to be exponential for the first 2 hours, and to go as $t^{-.21}$ thereafter. Thus, for the 2-hour dose the time factor, $G_i(\tau)$, is correct as given in eq. (31) for the solid fission products, but for longer exposures the expression must be modified. The integration over exposure time which gives the time factor is

$$\int_0^\tau \exp(-\lambda_i t) dt$$

For the long-term dose from the solid fission products we must replace this integral with the following expression:

$$\int_0^{7200} \exp(-\lambda t) dt + \exp(-7200\lambda) \int_{7200}^\tau t^{-.21} dt$$

This gives the time factor, $G(\tau)$, for the long-term solid fission product

dose as

$$G(\tau) = \frac{1 - \exp(-7200\lambda)}{\lambda} + \frac{\exp(-7200\lambda)}{.79} \left[\tau^{.79} - (7200)^{.79} \right], \text{ sec} \quad (32)$$

The absorption coefficients and time factors are given for each isotope in Table 14.6-4. The omission of data on KR-83^m is a result of the low gamma energy for this isotope, which yields a negligible contribution to the dose. The exposure time for the solid fission product group has been terminated at 30 days for the long-term case. This follows the practice of TID-14844 (14.7) and avoids the infinite dose which results from eqs. (32) and (29) for an infinite exposure time. The choice of a 30 day exposure time is entirely arbitrary, but should be long enough to carry out any evacuation measures which might be necessary because of the solid fission product activity.

The results of the direct gamma dose calculations (14.21) are given in Figure 14.16 for both the 2 hour exposure and the infinite exposure (the latter curve is marked "30 day" to emphasize that the solids dose is terminated at that point).

14.6.6 DISCUSSION OF RESULTS

14.6.6.1 The Overpressure Condition. The containment building overpressure which results from the maximum credible accident is shown in Figure 14.12. This illustration also shows the building atmosphere temperature during the period of the overpressure condition. The maximum overpressure is 0.90 psi and occurs about 1,000 seconds after the onset of the accident. The maximum atmosphere temperature is 128°F, at about 4200 seconds. The shape of the overpressure curve (Figure 14.12) is determined by the balance between the energy and vapor input to the building atmosphere (from the core afterheat and evaporation from the heavy water spill), and the energy and gas losses from the atmosphere (by heat transfer to concrete structures and leakage from the stack vent).

The building overpressure rises from an initial value of 0.1 psi, due to the assumed drop in barometric pressure in the outside atmosphere, to the maximum value of 0.9 psi and then drops to zero in a total elapsed time of 5000 seconds. The time-dependent nature of the overpressure results in time-dependent fractional leakage rates from the containment shell and the stack vent. The fractional leak rate curve for the stack vent is shown in Figure 14.13. The fractional leak rate curves for the building leakage alone, and for the sum of the building and stack leakages are similar in form to that shown for the stack and are obtained by multiplying the ordinate in Figure 14.13 by 0.00967 and 1.00967, respectively. All outward leakage ceases at 5000 seconds after the initiation of the accident when the building overpressure goes to zero. At very long times the building pressure will approach the negative gauge pressure provided by the draft of the stack. As noted in Section 14.4.5, the draft from the stack may be as little as 0.16 in. H₂O. The stack draft can, however, be much increased by the operation of the Graphite Reactor cooling air fans.

We have assumed for the maximum accident that the air conditioning system in the building is shut down. As noted in Section 14.6.2.3, if the air conditioning continues to operate during the accident (as one would expect), the building atmosphere temperature rise would be reduced. There would be a corresponding decrease in the overpressure in the first 5000 seconds, and the building atmosphere would come more rapidly to the slight negative gauge pressure dictated by the stack draft.

Table 14.6-4 Direct Gamma Dose Calculation Parameters

Isotope	β_{ie} (m^{-1})	β_i (m^{-1})	α_i	$G_i(\tau)$ 2 Hours (sec)	$G_i(\tau)$ Infinite (sec)
1. I-131	.0038	.0123	2.22	7199	101. x 10 ⁴
2. I-132	.0031	.0058	.9	5461	1.25 "
3. I-133	.0039	.0108	1.85	6973	10.8 "
4. I-134	.0034	.0072	1.10	3609	.455 "
5. I-135	.0033	.0067	1.02	6505	3.46 "
6. Kr-83 ^m	- -	- -	- -	- -	- - -
7. Kr-85 ^m	.0035	.016	3.5	6168	2.27 "
8. Kr-87	.0039	.0108	1.85	4426	.676 "
9. Kr-88	.0030	.0058	.9	5655	1.44 "
10. Xe-131 ^m	.0033	.017	4.0	7200	150. "
11. Xe-133 ^m	.0036	.015	3.2	7092	28.7 "
12. Xe-133	.0027	.022	7.0	7118	65.8 "
13. Xe-135 ^m	.0039	.011	1.9	1345	.135 "
14. Xe-135	.0036	.015	3.0	6682	4.74 "
15. Solids	.0038	.010	1.6	5634	*9.42 "

* 30 day exposure time

No excursion energy from a nuclear transient has been considered in the overpressure calculations. The assumptions which have been made for the maximum accident of gross failures of the reactor vessel and shield pit, of the maximum possible primary coolant spill, and of a full core meltdown encompass the worst possible effects from any nuclear transient which the HFBR could produce. The effect of adding the energy from a nuclear transient to that already considered in the accident is small. The type of transient of which the HFBR is capable will result in most of the energy of such an excursion appearing as temperature rise in the primary coolant. The addition of even several hundred megawatt seconds to the 4300 MW-sec which are the normal energy content of the full primary coolant inventory will have a completely trivial effect on the overpressure condition. The addition of nuclear transient energy directly to the atmosphere of the building will, of course, result in increased building pressures at earlier times. Such an energy addition would raise the initial overpressure to some value above the 0.1 psi barometric term, and the atmosphere temperature above the initial 70°F ambient. The greater initial overpressure would then result in leakage from the stack vent at a greater rate than in the case we have considered, and the higher air temperature would start the transfer of heat to the concrete structure at a greater rate and at an earlier time than in our calculation. The net effect of these factors is that the overpressure condition is terminated sooner, and that the net leakage of fission products is largely unchanged.

In the course of survey calculations on this problem we have calculated a case in which 1700 MW-sec were arbitrarily added to the conditions already assumed for the maximum credible accident. The total addition was divided into 500 MW-sec added to the primary coolant and 1200 MW-sec added to the building atmosphere. This survey calculation, of course, represents a totally arbitrary case and is in no sense to be regarded as a credible or possible accident. The results of this enormous addition of energy at the instant of the accident were a maximum overpressure of 1.72 psi at 400 sec, a maximum temperature of 145°F at 2200 sec, and a total overpressure duration time of 4100 sec. The total leakage of fission products from the building was only slightly larger than in the maximum accident and resulted in an increase in the peak thyroid doses of only 11% (14.18).

We have also considered cases in the survey calculations which lead to smaller doses than those reported here. In particular, the addition of heat transfer from the building atmosphere to the large surface of the steel dome, with all other conditions as in the maximum accident, results in a peak overpressure of 0.75 psi and a termination point for the overpressure condition of about 4000 seconds. The leakage in this latter case is, of course, less than that in the maximum accident and the doses are correspondingly decreased.

In summary, we feel that the conditions assumed for the maximum credible accident are already so severe in terms of the overpressure and gas leakage, and hence in terms of the cloud doses, that the addition of energy from any possible nuclear transient will make essentially no difference in the doses which we report.

14.6.6.2 Cloud Doses: The Exclusion Area, Low Population Zone, and Population Center Distance. The total thyroid and whole body gamma doses which result from the combined building and stack vent clouds are shown in Figures 14.14 and 14.15. The dose curves are shown for the four meteorological conditions which are considered important in the safety analysis (see Section 2.3.2).

The exclusion area is defined as that area around the reactor building at the outer limit of which a person will receive a total dose in the first

2 hours of the accident of no more than 300 rad to the thyroid, or 25 rad from whole body gamma irradiation (14.22). The total cloud dose from the maximum credible accident is experienced in less than 2 hours so the total dose curves are used to determine the exclusion area. Table 14.6-5 shows the distances from the reactor building at which doses of 300 rad to the thyroid and 25 rad from whole body gamma irradiation are received for each of the four meteorological cases.

Table 14.6-5 Exclusion Area and Low Population Zone Distances

Meteorological Case	Distance for 300 rad Thyroid Dose in First 2 Hours	Distance for 25 rad Gamma Dose in First 2 Hours
Case I	175 meters	<100 meters
Case II	175	<100
Case III	750	130
Case IV	<100	<100

The greatest distance listed in Table 14.6-5 is 750 meters, and the exclusion area distance for the reactor is accordingly set at 750 meters.

The low population zone is defined as that area around the reactor at the outer boundary of which a person would receive no more than 300 rad to the thyroid or 25 rad from whole body gamma irradiation for an exposure time which is infinitely long (14.22). Since the cloud doses are all received in less than 2 hours, this distance is identical with the exclusion area distance, 750 meters.

The population center distance is defined as 1.33 times the low population zone boundary distance (14.22). It is, therefore, 1.33×750 , or 1000 meters.

The nearest boundary of the BNL site is slightly more than 1100 meters from the HFBR building. Thus, the exclusion area, the low population zone, and the population center distance for the HFBR are entirely contained within the boundaries of the laboratory site. Further, within the site, the exclusion area and low population zone do not include the Medical Center, the hospital, nor the apartment area (see Figure 2.4).

The worst meteorological condition, in terms of doses of the magnitude of the guideline levels, is the strong inversion, Case III. This is the classical case for reactor safety analyses because of the poor diffusion and dilution in a cloud from a ground-level source. The inversion is a frequent occurrence during night hours in the Brookhaven area, and for this reason the building leakage in a major accident is much more serious than the stack release, even through the stack release is larger.

It is, in fact, just to reduce the severity of the building cloud dose in inversion conditions that we choose to vent from the stack in an overpressure condition. The stack vent has the useful function of reducing both the magnitude of the maximum overpressure and the duration of the overpressure condition. In turn, this substantially reduces the dose due to the inversion

case cloud from building leakage.

14.6.6.3 Direct Gamma Dose. The direct gamma dose from fission products which remain in the building is shown in Figure 14.16 for both the 2 hour exposure and for an infinite time exposure. The 25 rod dose in two hours occurs at 300 meters. The 25 rod dose in infinite time occurs at 450 meters. Both of these distances are less than the cloud dose distances, and lie well within the BNL site boundary. The direct gamma radiation from the building, therefore, constitutes no serious hazard to the general public. Note that the direct gamma dose from the building makes only a small (0.4 rad) contribution to the cloud gamma dose at the 750 meter exclusion area and low population zone distances.

References

- (14.1) 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).
- (14.2) G. W. Parker, private communication (submitted for publication in Nuclear Safety).
- (14.3) Pinson and Langham, "Physiology and Toxicology of Tritium in Man", Journal of Applied Physiology, 10, 108 (Jan., 1957).
- (14.4) P. R. Tichler, "Primary System Depressurization Due to Various Valve Openings in Off-Gas System", BNL Memorandum (Feb. 12, 1964).
- (14.5) P. R. Tichler, "Hot Spot Surface and Saturation Temperatures During Depressurization Accidents", BNL Memorandum (Feb. 27, 1964), and "Errata in Memorandum on Hot Spot Surface and Saturation Temperatures During Depressurization Accidents", BNL Memorandum (April 16, 1964).
- (14.6) K. C. Hoffman and T. V. Sheehan, "Utilization of No. 1 Research Reactor Stack for Proposed HFBR Building Ventilation Air", BNL Memorandum (July 6, 1959); M. E. Smith, "Backflow from the Research Reactor Stack into HFBR", BNL Memorandum (October 7, 1959); and T. V. Sheehan, "Stack Report--High Flux Reactor", BNL Memorandum (October 12, 1959).
- (14.7) J. J. DiNunno, F. D. Anderson, R. E. Baker, and R. L. Waterfield, Calculation of Distance Factors for Power and Test Reactor Sites, TID-14844, Div. of Licensing and Regulation, USAEC (March 23, 1962).
- (14.8) J. O. Blomeke and Mary F. Todd, Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time, ORNL-2127, Part I (TID-4500).
- (14.9) G. C. Kinne, "Summary of Fission Products for the HFBR", BNL Memorandum (April 21, 1961).
- (14.10) W. J. Megaw and F. G. May, "The Behaviour of Iodine Release in Reactor Containers", Reac. Sci. and Tech., 16, 427 (1962).

- (14.11) J. B. Morris, H. R. Diffey, B. Nicholls, and C. H. Rumary, "The Removal of Low Concentrations of Iodine from Air on a Plant Scale", Reac. Sci. and Tech., 16, 437 (1962).
- (14.12) Boelter, Gordon, and Griffith, Ind. Engr. Chem., 38, 596 (1946), as reported in Chemical Engineers Handbook, 3rd Edition, J. H. Perry, Ed. (McGraw-Hill, New York, 1950), p. 546, eq. (77).
- (14.13) K. Shure, WAPD-BT-24, 1-17 (December, 1961).
- (14.14) See, for example, Brown and Marco, Introduction to Heat Transfer (McGraw-Hill, New York, 1951), p. 136, or McAdams, Heat Transmission, 3rd Edition (McGraw-Hill, New York, 1954), Chap. 7.
- (14.15) H. S. Davis, Nucleonics, 13, 60 (1955).
- (14.16) "Report of ICRP Committee II on Permissible Dose for Internal Radiation (1959)", Health Physics Journal, 3 (June, 1960).
- (14.17) T. J. Burnett, Nucl. Sci. Engr. 2, 382 (1957).
- (14.18) L. G. Epel, "Radioactive Cloud Dose and the Maximum Credible Accident of the HFBR", BNL Memorandum (December 4, 1963), and "Radioactive Cloud Dose, etc., Erratum and Addendum", BNL Memorandum (Dec. 26, 1963).
- (14.19) L. S. Marks, Mechanical Engineer's Handbook, 5th Edition (McGraw-Hill, New York, 1951), p. 291.
- (14.20) H. Goldstein, The Attenuation of Gamma Rays and Neutrons in Reactor Shields (U. S. Gov't. Printing Office, Washington, May 1, 1957), p. 187, eq. (98).
- (14.21) L. G. Epel, "External Gamma Dose Subsequent to Maximum Credible Accident of HFBR", BNL Memorandum (Nov. 1, 1963).
- (14.22) "Reactor Site Criteria", Title 10, Code of Federal Regulations, Part 100 (10 CFR 100, February 11, 1961).

Figure 14.1 Rod withdrawal transient at 12.5 cents/sec from critical at an initial power of 10 KW. The shutdown is from 4 of the 8 main rods, with a 50 MW power trip level.

Figure 14.2 Rod withdrawal transient at 12.5 cents/sec from critical at an initial power of 40 MW. The shutdown is from 4 of the 8 main rods, with a 50 MW power trip level.

Figure 14.3 Central irradiation tube flooding transient from an initial power of 40 MW. The reactivity input is an 85 cent step. The shutdown is from 4 of 8 main rods, with a 50 MW power trip level.

Figure 14.4 Transient caused by starting a cold primary loop while operating at reduced power on one loop. The shutdown is from 4 of 8 main rods, with an overpower trip at 50 MW.

Figure 14.5 Transient from 10 KW initial power caused by 2g withdrawal of a broken auxiliary control rod. The total reactivity input is 1.12 dollars and the shutdown is from 4 of 8 main rods, with an overpower trip at 50 MW.

Figure 14.6 Transient from 40 MW initial power caused by 2g withdrawal of a broken auxiliary control rod. The total reactivity input is 1.12 dollars and the shutdown is from 4 of 8 main rods, with an overpower trip at 50 MW.

Figure 14.7 Pressure and temperature transients resulting from the accidental opening of valve HCe-102 with the reactor at full power.

Figure 14.8 Pressure and temperature transients resulting from the accidental opening of the 2-inch valve in line P118 with the reactor at full power.

Figure 14.9 Temperature transients resulting from the accidental functioning of the flow reversal system with the reactor at full power.

Figure 14.10 Total integrated thyroid dose from the fuel handling accident as a function of distance from the reactor stack.

Figure 14.11 Total integrated whole body gamma dose from the fuel handling accident as a function of distance from the reactor stack.

Figure 14.12 Overpressure and temperature of the reactor building atmosphere as a function of time after the maximum credible accident.

Figure 14.13 Fractional stack leak rate as a function of time after the maximum credible accident.

Figure 14.14 Total integrated thyroid dose from the maximum credible accident as a function of distance from the reactor.

Figure 14.15 Total integrated cloud whole body gamma dose from the maximum credible accident as a function of distance from the reactor.

Figure 14.16 Direct whole body gamma dose from the maximum credible accident as a function of distance from the reactor building.