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ARGONNE HIGH-FLUX RESEARCH REACTOR - AHFR
CONCEPTUAL DESIGN STUDY

by

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ARGONNE HIGH-FLUX RESEARCH REACTOR - AHFR CONCEPTUAL DESIGN STUDY

I. SUMMARY

This report presents a reactor design to meet the needs of the basic research program at Argonne National Laboratory. The program requires some irradiations in thermal neutron flux approaching $5 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ and some beam experiments having over $10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ available.

Parametric studies indicate that the highest peak thermal flux per unit of reactor power can be obtained in an internal H_2O reflector having a radius of about 6 cm. Since the design efforts recognized total reactor cost as a major parameter, an all H_2O -cooled system was adopted as most expedient. Further, the design and calculations effort was directed toward operating conditions which require little or no research effort. This latter point is a recognition both of the over-all cost plus a desire to meet the needs of research in the shortest possible time.

The reactor concept evolved is shown in Figs. 1-12. The design data are summarized in Table I.

Table I

SUMMARY OF AHFR DESIGN DATA

<u>Power</u>	
Heat, Mw	100
Heat Flux, Btu/(hr)(ft ²)	
Average	580,000
Maximum	980,000
Maximum to average, total	1.68
axial	1.29
radial	1.30
Power Density, Mw/liter	
Average	1.13
Maximum	1.90
Specific Power, Mg/kg fissionable material	23
Hot Channel Factors	
Bulk Rise	1.29
Film Rise	1.56

Coolant

Type	H ₂ O
Flow Rate, gpm	
Core	13,000
Reflector and Shield	2,000
Velocity in Core, ft/sec	35
Temperature, °F	
Inlet	135
Outlet	188
Pressure, psia	
Core Inlet	237
Core Outlet	200

Neutron Flux, n/(cm²)(sec)

	<u>Thermal</u>	<u>Fast</u>
Central Thimble (max.)	4.5	2.0
Core (avg)	1.3	3.0
Outer Reflector (max.)	1.6	2.2

Core

Geometry	Annular cylinder
Over-all Dimensions, in.	
Inside Diameter (square)	4.75 x 4.75
Outside Diameter (equiv. cylinder)	19.4
Height	20
Volume, liters	88.8
Fissionable Material, kg	~4.3
Fuel Elements	
Geometry	Flat plates
Clad material	Aluminum
Dimensions, in.	
Over-all	20.75 x 2.166 x 0.05
Active section	20 x 2.00 x 0.02
Heat Transfer Surface, in. ²	80.6
Uranium content, gm	~4
Fuel Subassembly	
Over-all dimensions, in.	26.5 x 2.359 x 2.359
Side Plate Thickness, in.	0.171
No. of Fuel plates	22
Water channels, in.	2.016 x 0.057
No. Subassemblies/core	48
Metal to water ratio (active zone)	1.15 : 1
Surface Temperature (max.), °F	315

Core (cont'd.)

Fuel Temperature (max.), °F	365
Burnup/cycle (avg.), %	~30
Fuel Cycle (avg.), days	10

Control

No. of Rods	8
Material	Boron-stainless steel (Beryllium follower section)
Burnable Poisons, cm ²	3,020
Excess Reactivity, %	
Burnup	11
Xe, Sm, fission products	6
Experiments	6

Reflector

Internal	H ₂ O
External	Beryllium
Thickness, in.	~12
Height, in.	~32

Vertical Experimental Facilities

<u>Type</u>	<u>Total</u>	<u>ID, in.</u>	<u>Location</u>
Central Thimble	1	1.5	Core Center
Irradiation Hole	10	1	1.25 in. outside core
Hydraulic Rabbits	2	1	1.25 in. outside core
Vertical Thimble	2	2	4.5 in outside core
	4	2	13 in. outside core

Horizontal Experimental Facilities

<u>Type</u>	<u>Total</u>	<u>ID, in.</u>	<u>Location</u>
Through Holes	2	6	5 in. outside core 13.5 in. below core center
Beam Holes	2	3	1 5 to 7 in. from
	4	4	core on core
	4	5	center plane

The cost estimate for the reactor facility is \$13,700,000. The annual operating cost, exclusive of fuel, is estimated at \$1,160,000. The fuel cost, excluding a burnup charge, is estimated to be of the order of \$600,000 and the annual quantity of U²³⁵ burned is 38 kg.

II. DESIGN DESCRIPTION

A. Reactor

1. Fuel Plate

The basic fuel plate (Detail 1, Fig. 1) is similar to that used in the ETR. The aluminum "meat" (0.020 in. thick) contains 4-5 gm highly enriched U^{235} and is clad with aluminum 0.016 in. thick. Each plate is 20.750 in. long, 2.166 in. wide, and 0.050 in. thick. The active length is 20 in., and the active width is 2 in.

2. Fuel Subassembly

Each fuel subassembly (Detail 8, Fig. 1) contains 22 fuel plates spaced by slots (0.075 in. deep) machined in two side plates (0.171 in. thick) to form coolant flow channels 2.016 by 0.057 in. The ends of the side plates are cross connected by 0.171 bridge members to form a box 2.359 in. square in cross section. The fuel plates are spaced at the upper and lower ends by a comb 0.125 in. thick. The upper end of the subassembly also features a hold-down bracket which may be used to facilitate fuel exchange.

3. Core Loading

A full core loading of 48 fuel subassemblies comprises four 6-unit assemblies (Detail 9, Fig. 1); four 4-unit assemblies (Detail 12, Fig 1); and eight single fuel subassemblies (Detail 8, Fig. 1). The total heat transfer surface is 587 ft². A fuel core loading is shown in Fig. 2.

4. Pressure Vessel

The reactor pressure vessel (Fig. 3) is 78 in. ID, 1.125 in. thick, 21 ft - 8.5 in. high, and made of Type 304 stainless steel. The lower extension (24-inch pipe, 82.5 in. long) contains the control rod drive shafts and the coolant inlet. The 24-inch diameter coolant outlet is located in the upper portion of the vessel. The vessel contains a 9-inch tube which is used for fuel transfer into the canal. At the upper end of the vessel is a flow barrier consisting of two semi-circular plates which are hinged at the outer sides. The barrier prevents a high concentration of N^{16} activity under the vessel cover. These plates are held down by triple-acting ball lock pins which can be unlocked to allow the two hinged plates to open outward to permit access to the core for fuel removal. There are a total of 10 penetrations in the pressure vessel for the horizontal beam holes which are located 42 in. from the floor line, thus placing them on the horizontal core center line. In addition, four penetrations are in the vessel wall 13.5 in. below the horizontal core center line. These are for the two through tubes. The ellipsoidal lower head contains two penetrations for the rabbit holes.

Equi-spaced radially around the core center line are six mounting lugs which nest into slots in the inner external thermal shield. The horizontal facilities are free from shear forces due to thermal expansion, since by so mounting, the pressure vessel is free to expand above and below this mounting point. At the upper end of the vessel centering springs provide for vertical alignment. At the lower end of the vessel, the support ring in the ellipsoidal head is centered in a guide ring welded to the bed plate. These additional guiding devices are shown in Fig. 4. Since the operating weight of the pressure vessel and attachments is about 58 tons, the unit load on the mounting lugs is about 9.5 tons or 2210 psi.

a. Core and Reflector Support Assembly

The core is supported by an aluminum grid (Section F-F, Fig. 3) which is bolted to a ring integral with the inner thermal shield. The grid section is basically a 0.344-inch by 2-inch bar, cross connected and welded to form a rigid support for the core. This grid in turn is corner connected to a heavier ring-shaped support by eight spacers 2 in. thick. The outer section members (2 in. wide x 0.75 in. thick) support the beryllium reflector. The entire unit can be removed for inspection and maintenance.

The upper end of the core is supported in a similar manner, but does not include the inner grid assembly (Section B-B, Fig. 3). It does include an aluminum plate (2 in. thick) with metering holes to control the coolant flow in the outer reflector. The top support unit is also mounted to the inner thermal shield.

b. Hold-Down Grid

The hold-down grid (Fig. 2) functions to hold fuel and reflector pieces in place and to transmit the forces generated on these pieces (by the upward flow of coolant) to the vessel walls. The grid consists of four hinged sections which press against the fuel upper support brackets. Each grid unit contains two triple-action ball lock pins which insure that the grid is locked in position during reactor operation. These pins are unlocked from the top of the pressure vessel during the fuel reloading cycle; the hinged hold-down grid can then be lifted outward to permit access to the fuel area. The hold-down grid hinge bracket is welded to the upper support and includes a down extension track which acts as a guide for the control plate and as a hold-down for outer reflector pieces.

c. Reflectors

There are two reflector zones: the inside water reflector which is contained centrally within a 4.375-inch square aluminum duct; and the beryllium reflector around the outer edges of the fuel boxes. This reflector is comprised of eight segments which are bolted together to form a cavity on the inside for the fuel assemblies.

5. Control Rods

Four plate-type and four angular-type control rods are located in the beryllium reflector (Fig. 2). All rods feature an upper poison section and a beryllium follower section. The surface areas are $\sim 0.50 \text{ ft}^2$ for the plate type, and $\sim 0.28 \text{ ft}^2$ for the angular rod. The actuating mechanisms are of the type used on the EBWR. As the poison section is driven upward out of the core, it is replaced with the beryllium follower. The scram action is downward, and is aided by spring force.

6. Thermal Shield

a. Inside Vessel

The internal thermal shield (Section A-A, Fig. 3) consists of two steel plates (1 in. thick) spaced 0.75 in. apart. The lower ends terminate in semi-dished heads containing a coolant flow opening (24 in. dia.). Around the periphery of these 24-inch diameter holes, spaced 120° apart, are three positioning notches which engage projections rising from the 2-inch thick combination transition ring located in the ellipsoidal head. This arrangement spaces the thermal shield sections both radially and vertically without restricting the thermal expansion. At the open upper end of each thermal shield section are four positioning lugs which nest into adjacent slots. The inside section nests in the outer shield, and the outer shield nests in a cavity on the inner wall of the pressure vessel. The inner thermal shield section is the foundation for the attachment of the upper and lower grid supports. The shield sections are replaceable. This feature applies to all internal structures.

b. Outside Vessel

The external thermal shield (Fig. 4) consists of two 2-in. thick stainless steel sections with 0.75-in. coolant channels between them. Both sections are contained in a stainless steel membrane (0.5 in. thick). The steel sections are embedded in the floor plate beneath the reactor vessel. The inner section serves as the main reactor support. The mounting lugs on the pressure vessel engage slots in brackets welded to the thermal shield, thus providing an accurate and rigid location point. A separate coolant circuit is used for the external portion of the thermal shield.

7. Biological Shield, Fig. 4

The biological shield is magnetite concrete (density = 220 lb/ft^3) $8\frac{1}{2}$ ft thick in the region extending from the ceiling of the sub-reactor room to a point 15 ft above the main floor. This region forms a thirteen-sided irregular polygon about 25.4 ft across flats. Above this

15 ft level, the shield is circular with a diameter of 21.1 ft and a height of 7.3 ft. The total volume of dense concrete shield above the sub-reactor room is 450 yd³.

8. Experimental Facilities

A total of 19 vertical, and 12 horizontal experimental facilities are provided in the reactor. The description of each facility is given in Table II; their arrangement is shown in Fig. 5.

Aluminum is used as the structural material wherever the facilities penetrate into the reflector. The wall thickness of all tubes is the minimum by the Code⁽¹⁾ in order to reduce gamma heating and neutron absorption.

Table II

EXPERIMENTAL FACILITIES

Facility No.	ID in.	Position - Degrees Clockwise from X-axis (See Fig. 5)	Distance from Vertical Center Line of Core in.	Neutron Flux Range $n/(\text{cm}^2)(\text{sec})(\times 10^{-14})$		Remarks
				Thermal	Fast	
VT-1	1.5	Center	Center	45	20	Mechanical Rabbit
VT-2	2	90	14	11	5	Thimbles terminate in beryllium reflector. Access to thimbles at top of reactor.
VT-3	2	270	14	11	5	
VT-4	2	80	19	5	1.2	
VT-5	2	100	19	5	1.2	
VT-6	2	260	19	5	1.2	
VT-7	2	280	19	5	1.2	
IH-1	1	39	10.5	25	15	No thimbles to top of reactor. Access to holes only during fuel change.
IH-2	1	54	10.5	25	15	
IH-3	1	80	10.5	25	15	
IH-4	1	100	10.5	25	15	
IH-5	1	126	10.5	25	15	
IH-6	1	219	10.5	25	15	
IH-7	1	234	10.5	25	15	
IH-8	1	260	10.5	25	15	
IH-9	1	280	10.5	25	15	
IH-10	1	306	10.5	25	15	
RH-1	1	141	10.5	25	15	Loading terminal in H ₂ O canal.
RH-2	1	324	10.5	25	15	
HB-1	5	0	10.5	25	15	
HB-2	4	26	16	8	3	
HB-3	4	45	11	15	11	
HB-4	3	119	11	15	11	
HB-5	5	152	16	8	3	
HB-6	5	180	10.5	25	15	
HB-7	4	208	16	8	3	
HB-8	3	239	11	15	11	
HB-9	5	301	16	8	3	
HB-10	4	332	16	8	3	
HTH-1	6	90	13	13	7	Center Line of Hole 11.5 in. below centerline of Core
HTH-2	6	90	13	13	7	

a. Center Thimble (VT-1)

The most important vertical facility is the central thimble (Fig. 6) which penetrates the inner H₂O reflector and thus provides the highest thermal flux available in the reactor, $\sim 4.5 \times 10^{15} \text{n}/(\text{cm}^2)(\text{sec})$. Its importance is further enhanced by a mechanical rabbit feature in the thimble to permit insertion and withdrawal of samples during reactor operation.

Access to the center thimble is from the top of the reactor through a lock incorporated in the removable access plug (Fig. 4). The sample to be irradiated is lowered through the lock by a long-handled tool and secured by a bayonet lock mechanism in the lower end of the retracted telescopic ball nut lead screw. The lead screw acts as a mechanical rabbit to lower the sample the remaining distance into the core.

The lead screw mechanism is driven by a variable speed electric motor electronically controlled as a part of reactor control system so that changes in reactivity caused by the slow sample insertion or removal are compensated by the regular control system. The exact position of the sample is known at all times. This is achieved by translating the revolutions of the lead screw to linear measurements on a position indicator in the control room. Accuracy of .0005 in. per foot of lead screw extension is possible through proper construction of the lead screw and selective sizing of the balls used in assembly.

Withdrawal of the sample from the core is accomplished by retracting the lead screw to the uppermost position and again using the sample transfer tool to unlock the sample tube from the mechanical rabbit. The sample is withdrawn from the rabbit into the lock in the removable access plug and is then transferred into a suitable coffin.

Cooling water for the sample is provided at all stages of the withdrawal operation. The water enters at the top of the thimble, flows down through the inside of the mechanical rabbit and sample tube, and returns up the annular space between the rabbit and the thimble wall. At the top thimble support, the cooling water is routed through passages in the support braces to pipes in the top shield plug and returned to the primary water system. The rate of cooling water flow is controlled by valving of the inlet line to give proper temperature regulation of the sample.

The lower end of the thimble is fabricated of aluminum tubing with a bolted transition to stainless steel tubing at a point above the top grid. Lateral supports for the thimble are located at the lower grid, the upper grid and baffle plate, and at the top of the thimble. The thimble is supported at the top by a flange which is bolted to the upper thimble

support. A thrust bearing, bolted to the flange on the thimble, supports the weight of the mechanical rabbit, allowing it to rotate during operation.

The lower section of the three-section telescoping lead screw is prevented from rotating by ball-bearing trunnions riding in grooves located on the inner surface of the steel section of thimble. The middle and upper sections of the lead screw are free to rotate in radial ball bearing support rings which also ride in the above mentioned grooves. The upper section of the lead screw is referred to as the torque tube (Section C-C, Fig. 6) since it transmits the power from the gear at its upper end to the ball nut at its lower end. During withdrawal the lower section retracts into the middle section until the stops engage, and then the middle section is prevented from rotating while it is drawn up into the upper section (or torque tube). The travel, in either direction, may be stopped and the position maintained by a magnetic brake on the motor drive shaft.

During periods when the mechanical rabbit is retracted, internal cooling is not provided for the lower end of the thimble; however primary cooling water orificed through the inner reflector is sufficient to prevent overheating of the aluminum section of the thimble.

b. Irradiation Holes (IH-1 through 10)

Ten of the twelve irradiation holes located in the beryllium reflector provide space in a relatively high thermal ($\sim 1.5 \times 10^{15} \text{ n/cm}^2 \text{ (sec)}$) for irradiation experiments of small volume.

These holes do not extend to the top of the reactor. Therefore, sample changes can be made only during reactor shutdown and the top access plug is removed. Figure 7 shows a typical irradiation hole and the manner in which the sample-containing tube is locked in position by a quick release bayonet-type connection. Zirconium springs at the lower end of the tube hold the tube in the locked position. These springs are in a zone of high flux and will suffer radiation damage. An nvt of 5×10^{21} is considered a safe limit for zirconium which would give a maximum life of about 10 weeks before the springs must be changed. This problem requires further investigation. If subsequent studies prove that the springs must be changed too frequently to be practical, the facility can be redesigned with the springs located in lower neutron flux regions.

A sample tube is cooled by the primary cooling water flowing through the sample tube and through an annular space between it and the walls of the irradiation hole. Proper orificing is necessary to provide adequate cooling depending upon the nature of the sample. This orificing is a part of the design of any irradiation sample and would be checked by preliminary flow tests before placing a sample in the reactor.

Sample tubes are inserted and removed with a ball pin-type handling tool similar to that used for transferring samples out of the center thimble (Fig. 6). The irradiated samples can then be dropped into the canal through the fuel chute or lifted into a coffin at the top of the reactor.

c. Hydraulic Rabbits (RH-1 and 2)

Two of the twelve holes in the beryllium reflector are used for hydraulic rabbits. They are connected by tubes (1 in. ID) through the bottom of the pressure vessel to terminals in the canal. The rabbits extend to the top of the active core providing space for several shuttles in the irradiation zone (Fig. 7).

The core terminals of the rabbit holes are equipped with spring and hydraulic shock absorbers to prevent damage to the shuttle or the tube. The shock absorbers are inserted and locked in place in the same manner as the irradiation sample containers and may be replaced if necessary, during fuel loading.

A means of indicating that the rabbit shuttle has arrived at the core requires further development effort.

The shuttle propulsion system is quite conventional. The shuttle is loaded into the canal terminal and, by proper valve arrangement, is carried by high-pressure water flowing in the tube up to the core. The velocity of the shuttle is regulated by the flow rate according to the weight of the sample. When irradiation is completed, the water flow is reduced until the weight of the shuttle overcomes the drag and it falls back down the tube into the canal to be recovered. The high-pressure water used to cool the rabbit discharges into the pressure vessel at the top of the core.

d. Vertical Thimbles (VT-2 through 7)

Six vertical thimbles which terminate in the outer reflector at an elevation approximating the bottom edge of the active core are accessible from the top of the reactor (Figs. 4 and 7).

Each thimble is an aluminum tube of constant inside diameter with a sealing flange in the top shield plug. Radial support is provided by braces located at the baffle plate and the top grid, and by spacers on the thimble in the outer reflector.

Each thimble is cooled by passing primary cooling water through the annular space between the thimble and the reflector. The sample within the thimble is cooled by auxiliary cooling circuits of either

gas or fluids as needed by the experiment. A typical experiment is one in which the sample is enclosed in a pressure-tight case that extends the full length of the thimble and contains its own coolant piping or heating element. Piping and leads extend through a flange at the top end of the vessel to utilities at the top of the reactor. The pressure case is loaded with the sample and sealed in the canal, transported in a suitable cask to the reactor top, lowered into the thimble, and the leads are connected. The canal loading is used since the pressure case is reusable and, thus, is active. After the irradiation is completed, the process is reversed and the sample is unloaded in the canal.

Experiment leads may be carried from the thimbles through trenches in the top of the reactor to instrumentation located as close as possible to the thimbles. Instrument cabinet pads are provided on the top of the reactor and on the balcony around the face of the biological shield. Utility outlets for the thimbles are located in receptacle boxes recessed in the floor around the top of the reactor near the instrument cabinet pads.

e. Beam Holes (HB-1 through 10)

Ten beam holes penetrate the reflector and terminate two to six inches from the fuel. Five of the beam holes have an unperturbed thermal flux of $\sim 1.5 \times 10^{15} \text{n}/(\text{cm}^2)(\text{sec})$. The remaining five holes have an unperturbed thermal flux of $\sim 6 \times 10^{14} \text{n}/(\text{cm}^2)(\text{sec})$.

Each beam liner is an aluminum tube supported by a bracket at the outer face of the beryllium, and by a bolted flange at the outer end of the aluminum liner in the biological shield. The right-hand side of Fig. 8 is typical of beam hole construction.

The aluminum beam hole liner is fastened by a gasketed bolted flange to the steel sleeve which extends from the pressure vessel outward to the flange connection. The liner is free to expand inward toward the core as it is not restrained by the support at the beryllium. All beam hole liners are replaceable.

Each beam hole is equipped with a motor-driven gate at the outer end of the liner tube, approximately half-way through the biological shield. This gate is constructed of steel and lead with cooling water passages. The front, side, and bottom surfaces of the gate are covered with boral to provide attenuation of the thermal neutrons; and a boral sheet is located between the steel and lead sections to capture neutrons thermalized in the gate.

In order to facilitate instrument leads from equipment in the beam holes, the bottom of the gate is slotted to provide a tunnel or

space for a conduit under the gate when it is closed. The tunnel extends back through the service plug to the port box at the reactor face.

The service plug (Fig. 8) provides access to the bolting flange for removal of the liner tube or other maintenance operations in the beam hole. The beam plug is provided as a supplement to the beam gate when work must be performed in the port box and is used only when the gate is lowered.

The port box furnishes a work space inside of the biological shield for certain types of equipment used in beam experiments. If shielding is required for activated materials or instruments in the port box, a lead gate may be lowered over the opening. This gate is motor-driven and controlled in the same manner as the beam gate.

As shown in Fig. 5, each beam hole is perpendicular to a flat face of the biological shield. The width of the faces varies from 4.75 ft to 7.5 ft. Located on each face, near the port box, are utility boxes where connections are available to the experimenters for the following:

- (1) Demineralized H₂O and drain
- (2) Compressed air - 100 psi
- (3) 120 volt, a-c, 30 amp, 1-phase
- (4) 240 volt, a-c, 30 amp, 3-phase
- (5) 110 volt, d-c, 20 amp
- (6) Two 2-inch conduits for special services for required (helium, nitrogen, vacuum, etc.)

Receptacle boxes with connections for the above utilities are available to beam hole at points around the walls of the reactor room opposite each face of the biological shield. Conduits connecting these boxes with those in the face of the biological shield are carried under the main floor to facilitate the wiring of beam experiments which extend to the wall.

f. Horizontal Through Holes (HTH-1 and 2)

Two horizontal through holes penetrate the outer reflector near the bottom edge of the core. These holes are parallel to the "Y" axis (Fig. 5) and are 13.5 in. below the horizontal center line of the core. Figure 8 shows the general arrangement from the face of the biological shield to the pressure vessel. Each hole is equipped with two water-cooled beam gates at either end of the aluminum liner tube and lead gates on the port boxes at the face of the biological shield.

The through-hole design is complicated by the necessity for an expansion joint at one end of the aluminum liner tube. Due to the difference in the coefficient of linear expansion between aluminum and

steel, plus the additional expansion of the liner because of its higher temperature in the vicinity of the core, some means of compensating for this expansion must be employed to prevent overstressing the aluminum tube. Expansion joints are used (Fig. 8) since (1) conventional bellows expansion joints are not feasible at the working pressure of 150 psi: the bellows would be too stiff to flex properly before the yield stress value for the tube is exceeded; and (2) "O" ring seals or packing glands are impractical because of the high radiation field.

Metallic piston rings provide a sliding seal between the liner and the sleeve, and by the use of six precision-fitted rings, a minimum leakage is expected. The leakage that does occur is non-active water and is carried off through the drain located in the seal. The seal is accessible for repairs during shutdown and the entire seal assembly can be removed and replaced.

g. Coffins

The normal operation of the reactor requires several special coffins for handling activated equipment and experiments. In order to reduce the number of coffins, it is desirable to design each to service as many different types of facilities as possible without sacrificing ease of handling. A universal coffin which is used to remove beam hole plugs, collimators, etc. from both the beam holes and through holes is possible due to the similarity in shape and size of this equipment. With minor modifications, a design similar to that of the universal coffin in use at the Materials Testing Reactor in Idaho should be very satisfactory.

A second coffin is necessary to transfer small diameter but long experimental apparatus from the vertical thimbles to the storage canal. This coffin could be simply a 4-inch I.D. tube approximately 20 ft. long covered with a tapered lead shield varying in thickness from about 11 in. at the bottom to 2 in. at the top, with a sliding gate at the bottom which can be opened remotely.

A smaller, but relatively more complicated coffin is necessary to handle samples irradiated in the center thimble. This is envisioned to be a lead cask approximately 2 ft in diameter and 5 ft high placed on top of the center thimble transfer lock into which the irradiated sample can be withdrawn. The coffin must be equipped with means of cooling the sample continuously during the removal operation and during transport to the destination.

Normally, samples removed from the irradiation holes (IH-1 through 10) will be passed through the fuel discharge chute into the canal and, therefore, will not require coffins for handling immediately on discharge from the reactor.

B. Cooling Systems

1. Primary System

The primary coolant system (Fig. 9) is designed to transfer 100 Mw of heat from the reactor core to the secondary system through conventional shell-and-tube heat exchangers. Most of the equipment is located exterior to the containment shell. Demineralized water at the rate of 15,000 gpm flows to the reactor shield at the basement level, rises and turns 180° near the top of the shield, and proceeds down to the inlet header at the bottom of the vessel. The top of the inlet water loop is vented into the reactor vessel to act as a siphon break. The main outlet is located at the same elevation as the top of the inlet loop so that water is retained in the reactor vessel even if the system integrity external to the biological shield is destroyed.

The outlet piping leads down through the shield to the basement level, through the containment building wall to a hold-up tank which allows the N^{16} activity to decay for 30 sec. This reduces considerably the amount of shielding required for the heat exchangers, pumps, degasifier, and demineralizing system. Table III lists data pertinent to the primary system components.

Table III

PRIMARY SYSTEM OPERATING DATA

Total flow, gpm	15,000
Total heat removal, Mw	100
Coolant temperature range, °F	135 - 188
Heat exchange surface, ft ²	16,000
System pressure drop, psi	65
Total pump power required, hp	800

Two 400-hp pumps operating in parallel against a head of 154 ft of H₂O supply the primary coolant to the reactor vessel. A large shell-and-tube heat exchanger is installed in each pump loop. The pumps and heat exchangers are cross connected so that any combination can be used. A 25-hp pump which bypasses the two main pumps supplies normal shutdown and emergency coolant to the reactor vessel. The shutdown pump motor features an emergency power supply.

The surge tank is at an elevation in the heat exchanger building such that its water height is the same as the minimum water level required in the reactor vessel during fuel transfers. This maintains the proper water level in the vessel when the gas pressure in the surge tank is atmospheric; i.e., the reactor vessel water level is not dependent upon

the pressurizing gas system to maintain a safe shutdown level. The pressurizing gas is admitted to the surge tank during normal operation, and a level control maintains the proper gas volume so that pressure fluctuations in the primary system are small.

The degassifier is operated at atmospheric pressure and admits 200 gpm of 188°F water which bypasses part of the main system (Fig. 9). This flow is pumped from the degassifier to a point where part of the flow is diverted to the demineralizing system. The remainder of the flow is pumped back to the main system. The pump used here supplies a constant 200 gpm back into the main system while the diversion allows a variable demineralizing flow. The valve which admits water to the degassifier controls the primary system pressure by allowing more or less water to flow from the system according as pressure is to be decreased or increased, respectively. Demineralized make-up water is added at the degassifier to compensate for system leakage.

The demineralizer flow passes through a heat exchanger, an ion-exchange column and a filter. Two stand-by columns are available for emergencies.

2. Secondary System

The secondary system supplies 25,100 gpm at 85°F to the heat exchangers. The heat is rejected to the atmosphere through a standard cooling tower which cools the water from 115°F to 85°F. Two main pumps require 240 hp each. A 25-hp pump supplies sufficient coolant to the heat exchangers for emergency situations. Because of evaporation (750 gpm) and blowdown (150 gpm), required make-up is 900 gpm.

3. Auxiliary Systems and Components

A storage tank (20,000 gal capacity) provides process water for demineralized make-up storage and for use in the emergency cooler-condenser.

Two additional cooling systems are provided for flow of demineralized water through the external thermal and biological shields and for use in removing heat from the experimental facilities.

C. Fuel Handling and Storage

1. Incoming Fuel

The fuel for the reactor is shipped to the reactor site fabricated into the final assemblies for loading into the reactor. These are stored in the fuel storage vault on the main floor of the containment building until placed into the reactor.

2. Loading and Unloading Operations

The fuel loading cycle is performed by first lowering the reactor system pressure to atmospheric pressure, then unbolting and removing the center plug and transfer lock assembly to the side of the reactor top decks. By means of a hand tool, the ball lock pins in the flow barrier are released and the flow barrier plates opened outward to permit access to the core. Since the fuel elements are clamped down by the fuel hold-down grids, it is necessary to unlock each of the four grids by tripping open the two-lock pins in each grid. These are not unlocked all at once, as one quadrant of the core is serviced at a time. A hand removal tool, having a long ball lock pin with an integral locating bracket, is inserted into the hole in the fuel assembly hold-down bracket and locked in position. The irradiated element is then lifted up out of the core and over to the fuel removal chute. The removal tool is disengaged allowing the element to settle into the lower lock where the transition to the canal is made. Each core quadrant is so emptied until the irradiated fuel is out of the core.

In loading the core, the following sequence must be followed. First the four 6-unit assemblies are inserted into the core cavity, the keys in these assemblies entering the slots in the square aluminum internal reflector shell and the opposite mating slots in the beryllium reflector. These assemblies, thus guided, are lowered into the core cavity and bottom in the lower support grid. The core cavity, after the insertion of these 6-unit assemblies, now contains the right size cavity to receive the four 4-unit assemblies which also bottoms in the lower support grid. Eight single subassemblies are inserted in the remaining core cavities to complete the loading (Fig. 2). The flow barrier is relocked, and the center plug and transfer lock assembly replaced and bolted in position to complete the reloading cycle.

3. Canal and Spent Fuel Storage

When the fuel assembly is lowered into the fuel removal chute the top valve is opened and the assembly enters the lock. The top valve is closed, and the bottom valve is opened allowing the assembly to move into a tilting basket attached to the chute. The assembly is then in a vertical position in the canal. A hand tool, operated from the main floor, is brought into position to pick up the assembly and move it to the fuel storage area of the canal. Forced coolant flow can be provided for the fuel assemblies during the entire transit time in the fuel discharge tube. The final design of the fuel tube will also include the provision for transferring materials from the canal, through the fuel discharge tube, and into the reactor.

The canal location is shown in Figs. 10 and 12. The canal is 50 ft long: 36 ft inside the containment shell and 14 ft outside. It is 6 ft wide and 21 ft deep, measured from the top of the parapet. The parapet

extends 3 ft above the first floor. The water level is 1 ft below the top of the parapet. Normal depth of the water in the canal is 20 ft. The section inside the containment building is common to the section outside by a 6 ft by 8 ft hole through the containment building wall at the bottom of the canal. It has a total area of 285 ft² on the bottom, 192 ft² inside the containment building, and 93 ft² beyond the containment wall. A 5-ton monorail crane operates over the canal. A conveyor on the canal insures the transfer of spent fuel and heavy items through the opening into the portion of the canal outside the reactor building.

With the normal operating cycle of the reactor, one spent core is stored about every 10 days. Additional fuel may have to be stored because of accidental shutdown of the reactor. With a cooling period of 100 days, at least ten core loadings may be stored in the canal. It is proposed that the spent fuel be passed through the opening in the containment building wall and stored in the section of the canal outside the containment shell. The total area necessary to store ten core loadings is approximately 15 ft². The storage space in this section of the canal equals 93 ft², or ample room for the normal load.

The first 9 ft of the canal, measured from the reactor face, are reserved for normal unloading of the fuel. In the case of an accidental reactor shutdown the fuel assemblies may be removed from the core and replaced in subsequent loadings. The reusable assemblies are stored in this area. The remaining 27 ft area of the canal inside the containment shell can be used for gamma studies and the storage of reactor equipment such as shroud, reflector parts, etc.

4. Spent Fuel Disposal

The spent fuel is removed from the canal at a station at the end of the canal in the equipment and heat exchange building. They are transferred from the canal into shipping coffins. The shipping coffins are stored in the storage area on the main floor of the equipment and heat exchange building. This arrangement of the transfer station of the H₂O canal being accessible to the storage area for the shipping coffins alleviates the over-all handling problems.

D. Buildings

1. Reactor Containment Building

The reactor containment building (Fig.12) is a reinforced concrete circular structure below grade and a steel dome structure above grade. It is 100 ft in diameter, 96 ft over-all height, with 27.5 ft below grade and 68.5 above. The gross area is 15,700 ft²; the gross volume is 761,800 ft³. The net usable area is 14,400 ft². The building has three working levels: the

basement, main floor, and the reactor top. The reactor is located centrally in the building and is serviced by an 80-foot span, 40-ton rotary crane.

The steel containment shell is 0.50 in. thick and is reinforced by internal stiffeners. The dome is designed to withstand a 1-psi positive differential pressure, and 2 in. of water negative differential pressure. Vacuum breakers insure that the 2 in. negative internal pressure is not exceeded. One personnel air lock, one equipment air lock, and one emergency air lock are at grade level. An equipment hatch is also provided for large equipment that cannot be handled by the equipment air lock. This hatch is used only during reactor shutdown. Other building penetrations include: the piping tunnel, the canal, electrical lines, and the ventilation and exhaust ducting.

a. Main Floor

The main floor (Fig. 10) is at grade level and is intended to serve as the experimental area. It contains a gross area of 7,850 ft² and a net usable area of 7,100 ft² with 3,840 ft² serviced by the 80-foot span rotary 40-ton crane. The floor is designed for a uniform loading of 1500 lb/ft².

The canal area, equipment hatch, heavy duty elevator, three stair wells, plug storage area, fuel storage vault, and emergency clothes change and shower room are located on the first floor. A staircase, above the canal allows access to the top area of the reactor.

b. Basement

The basement floor level (Fig. 11) is 18 ft below grade and has a ceiling height of 14 ft. The gross area is 7,850 sq ft and the net usable area is 7,280 sq ft. The floor in this area is designed for a uniform loading at 1500 lb/ft². The majority of the floor space is reserved for experimental equipment, with approximately 700 sq ft of shielded area set aside for storage of activated materials.

A portion of the monolithic canal shielding wall at basement level is replaced with removable shielding block. The resultant enclosure can be used for a dry irradiation facility which would receive its source of gammas from fuel elements assembled in the canal.

c. Control Room

The control room is located inside the containment shell and at the same elevation as the top of the reactor. An instrument room is located below the control room. The operators have a view of the reactor top and much of the working floor.

d. Storage

Additional shielded space is provided along the canal on the main floor for vertical storage of plugs. These long plugs can be lowered through a hole in the main floor into the canal shield. The close proximity of this area to the reactor reduces the handling time required for the removal and replacement of these plugs. Storage area for light equipment could be provided by constructing a loft perhaps at the 10 ft. level above the main floor in the area between the vertical support beams of the rotary crane and the wall of the containment building.

e. Utilities

In addition to the general operational utilities that are brought into the reactor building, each horizontal beam hole and the six vertical thimbles are supplied with demineralized water, compressed air (100 psi), and electrical power (110-v 30 amp a-c, 220-v 30 amp a-c, and 110-v 20 amp d-c). These utilities are placed at junction boxes on the wall of the reactor building near the center line of the horizontal beam holes. They also extend to the face of the reactor and to the reactor top through individual conduit brought through the utility openings in the main floor at the wall and reactor face. Additional power for the basement area can be supplied from the junction boxes on the main floor through the utility openings in the main floor.

2. Equipment and Heat Exchange Building

In order to keep a major portion of the reactor building for experiments, the primary heat exchange system and auxiliary equipment is placed in an adjacent structure (Figs. 10 and 12). The two buildings are connected by a pipe tunnel, the canal and an equipment air lock. The pipes are in the tunnel seal in the wall at the building line between the two structures (Fig. 12). The canal is extended through the containment building by a 6 ft by 8 ft hole in the wall of the containment building. The normal water depth of the canal is 20 ft. With this design, there is 12 ft of water above the upper edge of the hole in the containment wall to provide an adequate seal for differential pressure. The building is 56 ft high, 62 ft wide, and 88 ft long. The gross volume is 306,000 ft³. The basement and the main floor levels have a gross area of 12,200 ft² and a net usable area of 9,280 ft². A heavy-duty freight elevator, personnel and service entrances, and a 35-ton bridge crane are provided.

a. Basement

The basement floor level is 15.5 ft below grade level with a 13 ft ceiling height. The gross area is 5,500 ft² with a net usable area of 4,950 ft² and a normal floor loading of 500 lb/ft².

Figure 11 shows the arrangement of the heat exchange and water processing equipment essential to reactor operation. The primary heat exchange equipment consists of two heat exchangers and two pumps. Each component is located in a shielded cubicle to facilitate maintenance or repair during reactor operation. Other systems that present a radiation hazard, i.e., external shield cooling system, the experimental facilities cooling system, and the primary cooling clean-up system are located behind a common shielding wall.

The balance of the equipment in the basement, i.e., the secondary cooling system pumps and the demineralizing make-up system do not present a radiation problem; therefore, no shielding is necessary.

b. Main Floor

The main floor (Fig. 10) is at grade level. It has a ceiling height of 37 ft, a gross area of 5,500 ft², and a net usable area of 4,620 ft². The area has two levels: the main floor and the balconies along two sides.

Auxiliary and emergency equipment for safe reactor operation and shutdown are placed under the balcony. This equipment consists of (1) an emergency battery to supply d-c to the reactor and the experiment instrumentation in the event of a power failure; (2) a diesel generator set for emergency power; and (3) two air compressors for normal as well as emergency operation. The area under the balcony contains the heating and ventilation system auxiliary equipment, locker room, shower room, toilet, and office space. The ventilation system blowers and additional locker, toilet, and office space are on the balcony floor.

A storage area for shipping casks and noncontaminated material as well as machine shop area are placed on the floor in the space not influenced by the heat exchange cubicle covers. The area influenced by the covers lends itself to a traffic aisle and mock-up area. All areas, with the exception of the balcony area, are accessible to the crane. A health physics office is located beside the air lock to facilitate inspection of personnel and equipment.

3. Administration Building

The administration building (Fig. 10) is 128 ft long, 42 ft wide, with an over-all height of 27 ft (13 ft below and 14 ft above grade level), with a gross volume of 131,000 ft³, and gross area of 10,080 ft² (main floor and full basement). The ceiling height in the main floor is 9 ft, with a 3.5 ft high fan and service loft. The ceiling height in the basement is 12 ft. A personnel air lock leads from the administration building to the main floor of the reactor building.

a. Main Floor

The main floor is divided into two wings in an effort to isolate the administrative function from the physical operation of the reactor. The wing for the operating staff contains sanitary facilities, stock room with shipping area, two offices, and a health physics office adjacent to the personnel air lock.

The other wing contains the women's rest room, reception office, four additional offices for the operating staff, conference room, and storage space for office equipment.

b. Basement

The basement contains the building auxiliary equipment, i.e., air conditioning, ventilation fans, etc., building maintenance shop, and storage for equipment to be used on the reactor operating floor.

4. Miscellaneous Structures

The miscellaneous supporting structures include:

(1) A high ventilation stack and integral stack building containing high-efficiency filters to prevent contamination of surrounding terrain by air-entrained radioactive particulate.

(2) Cooling towers to dissipate the 100 Mw (heat) generated by the reactor.

(3) Retention lagoons for holding up and checking the blowdown water of the cooling towers.

(4) A 20,000 gallon emergency cooling water tower.

III. PHYSICS

A. Criteria

The physics calculations are based on an idealized system which is not identical with the AHFR design concept. The differences, however, are not sufficient to alter the final conclusions.

The basic criteria used in the design of the internal reflector-core-outer reflector system are as follows:

- (1) The peak unperturbed thermal neutron flux in the internal reflector should be at least $3 \times 10^{15} \text{ n/(cm}^2\text{)(sec)}$ for the loaded core at an average power generation of about 1 Mw/liter.
- (2) The materials used should be of relatively low cost and should require a minimum of research and development effort. In reference to this criterion, it was decided that the fuel elements should be Al- U^{235} plates and the internal reflector, core coolant, and core moderator should be H_2O .
- (3) The metal-to-water ratio of the core should be approximately unity in order to insure an adequate quantity of metal for heat transfer surface.
- (4) Total reactor power should be as low as possible consistent with (1).

B. Core - Reflector Studies

1. Optimization Study

An optimization study was made to determine what combinations of materials and configurations would satisfy the design criteria with the highest efficiency, i.e., highest peak thermal flux per megawatt. The cross sections employed are described on page 31. The parameters varied in this study were.

- (1) Core: volume, height, thickness, metal-to-water ratio, and U^{235} concentration.
- (2) Internal reflector: radius, effect of capping with Be plugs.
- (3) Outer reflector: respective thickness of the following combinations of materials: Be; Be followed by graphite (C); and H_2O followed by graphite.
- (4) Buffer: thickness and composition (Be, C, Mg or Pb).

The buffer region is composed of a good scattering but not necessarily good moderating medium located between the internal reflector and the inner surface of the core. The buffer was inserted so that fast neutrons generated in a thin, heavily loaded core of large mean radius (and hence large volume) would be transmitted without much loss to an internal reflector of optimum size, to avoid excessive loss of thermal neutrons by capture in H_2O .

Problems representing the various configurations were programmed for UNIVAC computation (RE-6 Code).⁽²⁾ This one-dimensional code iterates for the critical core composition using a given set of dimensions and compositions for regions other than the core. In addition to the critical composition, the code gives the radial flux distributions and the power distribution. The total axial reflector savings were taken to be 15 cm. Table IV lists the results of these problems (critical mass, critical composition, peak thermal neutron flux/Mw, and peak thermal flux at 1 Mw/liter) along with a complete description of the problem.

2. Optimum Core

Some conclusions that can be drawn from the results in Table IV and the associated flux distribution are:

- a. The presence of a buffer region between the core and the internal reflector appreciably lowers the peak thermal flux in the internal reflector.
- b. With no buffer region, the peak thermal flux is lowered when the radius of the internal reflector is in excess of 6 cm.
- c. The peak thermal flux and critical mass are insensitive to changes in the height-to-thickness ratio at a constant core volume of approximately 77.5 liters.
- d. The peak thermal flux and critical mass increase slightly with the metal-to-water ratio.
- e. The particular reflector configuration used has a small effect on the peak flux, but does have a large effect on the critical mass and the thermal flux peaking in the outer reflector. The three most promising configurations, in order of merit, are:

- (1) 30 cm of Be
- (2) 5 cm of Be followed by 40 cm of graphite. The thermal flux peak in the reflector is down $\sim 15\%$ from case (1)
- (3) 5 cm of H_2O followed by 40 cm of graphite. The thermal flux peak in reflector down 15% and critical mass up 1 kg from case (1).

TABLE IV
SUMMARY - CORE OPTIMIZATION STUDY

DESCRIPTION											RESULTS				
Internal Reflector Radius (cm)	Buffer Material	Buffer Thickness (cm)	Core			Metal to Water Ratio	First Reflector		Second Reflector		Critical Mass kg of U^{235}	Critical Concentration kg/l of U^{235}	Peak Flux per MW ($cm^{-2}sec^{-1}$) $\times 10^{13}$	Peak Flux at 1MW/liter $\times 10^{15}$	Prob. No.
			Thickness (cm)	Height (cm)	Volume (liters)		Composition	Thickness (cm)	Composition	Thickness (cm)					
5	C	10	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	30		None	2.451	0.0513	4.281	2.043	28
5	C	7	11	50	60.48	1 1	0.9 Be 0.1 H ₂ O	30		None	2.293	0.0379	4.350	2.631	33
6		None	11	50	39.74	1 1	0.9 Be 0.1 H ₂ O	30		None	1.8928	0.0476	6.894	2.740	8
6		None	17	50	77.44	1 1	0.9 Be 0.1 H ₂ O	30		None	2.3151	0.0299	4.571	3.540	9
8		None	15	53	77.42	1 1	0.9 Be 0.1 H ₂ O	30		None	2.6285	0.03395	3.827	2.963	38
6		None	19	42	77.72	1 1	0.9 Be 0.1 H ₂ O	30		None	2.4123	0.03104	4.513	3.508	39
6		None	15	61	77.61	1 1	0.9 Be 0.1 H ₂ O	30		None	2.3082	0.02974	4.543	3.526	40
6		None	17	50	77.44	9 11	0.9 Be 0.1 H ₂ O	30		None	2.1745	0.02808	4.414	3.418	41
6		None	17	50	77.44	11 9	0.9 Be 0.1 H ₂ O	30		None	2.5044	0.03234	4.739	3.670	42
6		None	17	50	77.44	1 1	0.9 Be 0.1 H ₂ O	5		None	2.4045	0.03105	4.631	3.586	43
6		None	17	50	77.44	1 1	0.9 Be 0.1 H ₂ O	5	0.9 C 0.1 H ₂ O	40	2.4440	0.03156	4.611	3.571	44
6		None	17	50	77.44	1 1	0.9 Be 0.1 H ₂ O	5	0.9 C [†] 0.1 H ₂ O*	40	2.3738	0.03055	4.588	3.552	45
6		None	17	50	77.44	1 1	0.9 Be 0.1 H ₂ O	10	0.9 C [†] 0.1 H ₂ O	35	2.3413	0.03023	4.580	3.547	46
6		None	17	50	77.44	1 1	H ₂ O	5	0.9 C [†] 0.1 H ₂ O	40	3.3021	0.0426	4.482	3.472	47
6		None	17	50	77.44	1 1	H ₂ O	10	0.9 C [†] 0.1 H ₂ O	40	3.5876	0.0463	4.494	3.480	48
6		None	17	50	77.44	1 1	H ₂ O	20	0.9 C [†] 0.1 H ₂ O	30	3.6356	0.0469	4.502	3.486	49
6		None	17	50	77.44	1 1	H ₂ O	10	0.9 C 0.1 H ₂ O	40	3.5945	0.0464	4.495	3.481	50
6		None	19	42	77.72	1 1	0.9 Be 0.1 H ₂ O	5	0.9 C 0.1 H ₂ O	40	2.5441	0.0327	4.568	3.550	52
5	0.9 Be 0.1 H ₂ O	10	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	30		None	2.5266	0.0529	3.393	1.620	1
4	0.9 Be 0.1 H ₂ O	8	11	50	60.48	1 1	0.9 Be 0.1 H ₂ O	30		None	2.2826	0.0377	4.043	2.445	2
6	0.9 Be 0.1 H ₂ O	9	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	30		None	2.6233	0.0549	3.064	1.463	3
5	0.9 Be 0.1 H ₂ O	7	11	50	60.48	1 1	0.9 Be 0.1 H ₂ O	30		None	2.3600	0.0390	3.757	2.272	4
5	0.9 Be 0.1 H ₂ O	7	14	50	83.57	1 1	0.9 Be 0.1 H ₂ O	30		None	2.6169	0.0313	3.116	2.604	5
5	0.9 Be 0.1 H ₂ O	4	8	50	32.67	1 1	0.9 Be 0.1 H ₂ O	30		None	1.9726	0.0604	6.218	2.052	6
5	0.8 Be 0.2 H ₂ O	10	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	30		None	2.6973	0.0565	3.161	1.509	7
5	0.9 Be 0.1 H ₂ O	10	11	50	70.84	1 1	0.9 Be 0.1 H ₂ O	30		None	2.6858	0.0379	2.783	1.972	10
5	0.9 Be 0.1 H ₂ O	10	8	35	33.43	1 1	0.9 Be 0.1 H ₂ O	30		None	2.9504	0.0883	3.941	1.318	11
5	0.9 Be 0.1 H ₂ O	10	8	65	62.10	1 1	0.9 Be 0.1 H ₂ O	30		None	1.7976	0.0289	3.290	2.041	12
5	0.9 Be 0.1 H ₂ O	10	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	20		20	2.5399	0.0532	3.393	1.620	35
5	0.9 Be 0.1 H ₂ O	10	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	10	C	30	2.6477	0.0554	3.394	1.621	36
5	0.9 Be 0.1 H ₂ O	10	8	50	47.75	1 1	C	40		None	3.0939	0.0648	3.463	1.654	37
5	Mg	10	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	30		None	4.1676	0.0873	3.376	1.612	21
4	Mg	11	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	30		None	4.1127	0.0862	2.982	1.424	22
6	Mg	9	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	30		None	4.2581	0.0892	3.6593	1.747	23
5	Mg	7	11	50	60.48	1 1	0.9 Be 0.1 H ₂ O	30		None	2.9574	0.0489	3.5812	2.160	24
5	0.9 Mg 0.1 H ₂ O	10	8	50	47.75	1 1	0.9 Be 0.1 H ₂ O	30		None	3.6239	0.0759	3.5642	1.702	26
6	0.9 Pb 0.1 H ₂ O	4	13	50	67.38	1 1	0.9 Be 0.1 H ₂ O	30		None	2.3261	0.0345	4.0011	2.696	83

*C[†] is high density graphite ($\rho = 1.9 \text{ gm/cm}^3$)

f. Nothing is gained by using high-density graphite instead of ordinary graphite.

The effect on reactivity and peak flux of beryllium plugs limiting the height of the H₂O internal reflector was also investigated. Two problems were run on the IBM-704 (using the two-dimensional PDQ Code⁽³⁾) which were identical except that in one problem, 10 cm thick beryllium plugs above and below H₂O internal reflector limited its length to 30 cm. The results showed the presence of the Be plugs increased the reactivity of the system about 1% and the peak thermal flux about 6%.

The composition and configuration of the core-reflector system which satisfies the design criteria and provides the greatest peak thermal neutron flux per megawatt is tabulated below:

Inner Reflector

Radius	6 cm
Height	30 cm H ₂ O with top and bottom plugs (10 cm thick) containing 90% Be, 10% H ₂ O

Core

Height	50 cm
Thickness	17 cm
Volume	77.44 liter
Clean critical mass	2.30 kg U ²³⁵
Metal:Water ratio	11:9

Outer Reflector

Thickness	30 cm
Composition	90% Be - 10% H ₂ O

C. Cross Sections

The 4-group cross section set used in the parametric and criticality studies are listed in Table V. The cross sections of U²³⁵, Al, and Be are the data of Avery, et al.;⁽⁴⁾ the water cross sections were reported by Hummel, et al.⁽⁵⁾ The cross sections for C, Mg, and Pb were computed from BNL-325,⁽⁶⁾ in the following manner:

Step 1

$$\tau(u_1, u_2) = 1/3 \int_{u_1}^{u_2} du \frac{g(u)}{\xi \Sigma^2(1-u)}$$

where

$$g(u) = \int_{\infty}^u F(u') \, du' + F(u)$$

$F(u)$ = fission spectrum.

Step 2

$$\text{Compute: } \int_{u_1}^{u_2} du \frac{g(u)}{\xi \Sigma_s}$$

Step 3

$$3 \Sigma_{tr} = \int_{u_1}^{u_2} du \frac{g(u)}{\xi \Sigma_s \tau(u_1, u_2)}$$

Step 4

$$D = \frac{1}{3 \Sigma_{tr}}$$

Step 5

$$\begin{aligned} \Sigma_{\text{removal}} &= \xi \Sigma_{tr} / \Delta u \text{ in the highest group (to allow for forward} \\ &\quad \text{scattering)} \\ &= D / \tau \text{ in groups 2 and 3.} \end{aligned}$$

The choice of Δu for the top group is somewhat arbitrary; a lower limit for u corresponding to an upper energy of 10 Mev was taken. The fission spectrum was taken from the tabulations in ANL-5800.⁽⁷⁾

Thermal group cross sections were computed assuming both core temperature and neutron temperatures to be 20°C. The concentration of an ideal $1/v$ absorber was varied for burnable poison studies, with its macroscopic absorption cross section in the fourth group arbitrarily normalized to 2.5 cm^{-1} . Similarly, parameters were calculated for a hypothetical material representing equilibrium xenon and samarium, and corresponding fission product poisoning. For a given situation the required concentration of this material could be computed.

D. Burnup and Control Calculations

Constant power operation at 1 Mw/liter ($\sim 80 \text{ Mw}$) corresponds to a burnup of approximately $100 \text{ gm U}^{235}/\text{day}$. Since the core is relatively heavily loaded, the variation in peak thermal flux with burnup is expected to be small so that operation at constant power is nearly equivalent to operation at constant peak flux.

Table V

FOUR-GROUP CROSS-SECTION SET

Energy Group	1	2	3	4
Lower Energy, E_L	1.353 Mev	9.12 kev	0.4 ev	0
Lethargy, $U = \ln \frac{10}{E_L(\text{Mev})}$	2	7	17.03	-
Fission Spectrum	0.575	0.425	0	0
<u>U²³⁵</u>				
ν	2.65	2.55	2.5	2.5
σ_{tr} , barns	4.7	7.0	5.1	597
σ_f	1.3	1.4	23.0	490
σ_c	0.1	0.3	18.0	97
$\sigma_{j \rightarrow j+1}$	1.4	0	0.01	0
<u>H₂O</u> (Molecule)				
σ_{tr}	3.08	10.52	16.55	68.6
σ_c	0	0	0.035	0.57
$\sigma_{j \rightarrow j+1}$	2.81	4.04	4.14	0
<u>Be</u>				
σ_{tr}	1.53	4.0	5.56	6.5
σ_c	0.06	0	0	0.009
$\sigma_{j \rightarrow j+1}$	0.47	0.21	0.088	0
<u>C</u> ($\rho = 1.60$)				
σ_{tr}	1.485	3.253	4.4	4.6
σ_c	0	0	0	.0045
$\sigma_{j \rightarrow j+1}$	.9316	.1269	.0738	0
<u>Al</u>				
σ_{tr}	1.8	3.0	1.4	1.6
σ_c	0.0003	0.0024	0.01	0.20
$\sigma_{j \rightarrow j+1}$	0.38	0.002	0.008	0
<u>Mg</u>				
σ_{tr}	1.222	4.3	3.26	3.26
σ_c	0	0	0	.069
$\sigma_{j \rightarrow j+1}$	.05	.07	.027	0
<u>Pb</u>				
σ_{tr}	3.5	6	11	11
σ_c	0	0	0	.17
$\sigma_{j \rightarrow j+1}$	.4	0	0	0
Poison ($1/\nu$) Σ_c , cm^{-1}	0	.0000945	.0018132	2.5
Poison (Xe + Sm) Σ_{tr} , cm^{-1}	0	0	0	17.676
Σ_c , cm^{-1}	0	0	0	5.892

Arbitrarily, 6% reactivity is assigned to experiments and about 6% to poison such as Xe^{135} , Sm^{149} and non-saturating fission products. To compensate for these effects, about 48% of the cold, clean critical mass must be added to maintain criticality. This amounts to about 1100 gm. Arbitrarily, 900 gm more has been added to give a cycle of about 10 days and an initial loading of 4.3 kg. Subsequent calculations show that at least 1100 gm can be burned up before the reactor becomes subcritical (~ 9 days at 100 Mw).

It is proposed that the initial loading be increased by 400 gm to 4.7 kg to give a cycle of about 14 days including reloading time. Since the Xe and other poisons correspond to ~ 650 gm, the proposed initial loading of 4.7 kg will require substantially less control than in the core of Mighty Mouse⁽⁸⁾ (although the total burnup is about the same), since the clean, cold critical mass plus Xe compensation is almost double that in Mighty Mouse.

Control by a sheet of rods between the core and the first outer reflector is possible. A thermally black sheet at the interface compensates 15.5% k in a core loaded with 4.3 kg and a sheet black only to epithermal compensates 8.5% k. Since only 11% is needed initially (and then only for a short time) if there is 6% in experiments, it follows that rods at the interface are sufficient. Unfortunately such rods yield a maximum to average power ratio greater than 4:1 and must be rejected on this basis. Fine control can be accomplished by rods in the outer reflector.

Control by burnable poison can be accomplished by the insertion of 3,020 cm^2 of thermal poison uniformly through the core for an initial loading of 4.3 kg; 178 cm^2 represent 1% in reactivity on this basis, but the worth per cm^2 depends upon how much poison has already been added. The maximum to average power ratio is 1.68 (to be compared with 3.28 for a clean, cold core); thus the burnable poison yields a favorable flux distribution. There is some evidence that the shift in the flux distribution tends to increase the vertical reflector savings which is an unexpected reactivity bonus. The peak flux at the 4.3 kg loading is $3.89 \times 10^{13} \text{ n}/(\text{cm}^2)(\text{sec})/\text{Mw}$, or $3 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ at 1 Mw/liter.

After 1000 gm of the original 4.3 kg have been burned up only 494 cm^2 of burnable poison are required at 84 cm^2 for 1% in k. This is slightly less than the 6% required for experiments; but the reactivity held in Xe, Sm, and other fission products has probably been over-estimated so that about 200 gm more burnup may be obtained. At this point the maximum to average power ratio is 2.32, nearly that of the cold, clean core. The peak flux/Mw is $4.15 \times 10^{13} \text{ n}/(\text{cm}^2)(\text{sec})$, or a peak flux of $3.21 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ for 1 Mw/liter. The change to an initial loading of 4.7 kg should not effect a significant change in flux or reactivity at 1000 gm burnup (3.3 kg in core) since Xe and Sm poisons are at, or near, saturation, and the contribution from non-saturating fission product poisons is small.

Figures 14 to 17 show the thermal flux distribution (PDQ data) in a core loaded initially to 4.3 kg, and for a core containing 3.3 kg of U^{235} with about 1 kg of fission products (including Xe^{135} and Sm^{149}). The fluxes have been normalized to one at the origin for convenience. In general, the epithermal (group 3) flux tends to follow the shape of the thermal flux but with less over-all variation. The fast fluxes (groups 1 and 2) are almost flat in the core and fall off at nearly exponential rates in the inner and outer reflectors.

E. Shielding

1. During Operation

The following radiation levels for the areas shown in Table VI have been specified.

Table VI

DESIGN CRITERIA

<u>Location</u>	<u>Allowable Radiation</u>
Inner-face of pressure vessel	1.8×10^{13} Mev/(cm ²)(sec), total gamma flux
Inner-face of concrete	2×10^{11} Mev/(cm ²)(sec), total gamma flux
Experimental areas on main floor	10 Mev/(cm ²)(sec), total gamma flux 1 n/(cm ²)(sec), total neutron flux regardless of energy
Canal (containing 3-hour decay fuel)	
Water surface	100 Mev/(cm ²)(sec)
Sides below main floor	500 Mev/(cm ²)(sec)
Sub-reactor room during operation	100 mr/hr
Top of reactor during refueling	4 mr

The shielding requirements for the core are ascertained by considering the bulk shielding to be divided into three general areas: the radial shield, the area below the core, and the space above the core.

a. Radial Shield

Calculations of the core and capture gamma fluxes and the neutron fluxes for various shield configurations result in a radial shield design which satisfies the conditions listed in Table VI.

The core gamma flux calculations are based on the following conditions:

Core radius	- 23 cm
Core height	- infinite (actual 50 cm)
Power density	- 1.2 Mw/liter
Aluminum-to-water ratio	- 1
Average thermal neutron flux	- $1.5 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$

The gammas originating in the core are produced mainly as a result of three separate processes: thermal neutron capture by aluminum, radioactive decay of fission products, and the fissioning of U^{235} .

Equation (1) is used to determine the gamma flux, $\Phi\gamma(0)$, in $\text{Mev}/(\text{cm}^2)(\text{sec})$ at the horizontal center line of the core surface for 5 energy groups: 1, 2.5, 4, 6, and 7.7 Mev.

$$\Phi\gamma(0) = \frac{S_v(E)}{2u_s} \quad (1)$$

where

$S_v(E)$ = volumetric source strength at energy E, $\text{Mev}/(\text{cm}^3)(\text{sec})$

u_s = gamma linear absorption coefficient of the core, cm^{-1}

The core gamma flux at the region interfaces is given by:

$$\Phi\gamma(x) = \Phi\gamma(0) \sum_{i=1}^i B_i \sqrt{r/r_0} E_i \left(\sum_{i=1}^i u_i t_i \right) \quad (2)$$

where

$\Phi\gamma(0)$ = gamma flux at the core surface, $\text{Mev}/(\text{cm}^2)(\text{sec})$

B_i = buildup factor for the i^{th} shield

r = radius of core, cm

r_0 = radius of point from center of core, cm

u_i = gamma linear absorption coefficient for i^{th} region, cm^{-1}

t_i = thickness of i^{th} shield, cm

E_i = function as defined in Ref. 9

The fast neutron flux at the inner face of the thermal shield was estimated at $3 \times 10^{11} \text{ n}/(\text{cm}^2)(\text{sec})$. The fast neutron removal cross section (Σ_i) is substituted for u_i in Eq. (2) and the buildup term is deleted.

Using slab geometry, the thermal flux in the shields beyond the beryllium is given by:

$$\phi_{th}(x) = A_i e^{\kappa_i x} + B_i e^{-\kappa_i x} + C_i e^{-\sigma_i x} \quad (3)$$

where

A_i, B_i = constants determined by the boundary conditions for the i^{th} region

$$C_i = \frac{\sigma_i \phi_f(0)}{D_i (\kappa_i^2 - \sigma_i^2)} \quad \text{n}/(\text{cm}^2)(\text{sec})$$

σ_i = slope of fast flux in the i^{th} region, cm^{-1}

D_i = thermal diffusion coefficient, cm

$$\kappa_i^2 = \Sigma_i / D_i, \text{ cm}^{-2}$$

Σ_i = thermal absorption cross section, cm^{-1}

$\phi_f(0)$ = fast flux at beginning of region i , $\text{n}/(\text{cm}^2)(\text{sec})$

The thermal flux at the inner face of the thermal shield was estimated at $6 \times 10^{11} \text{ n}/(\text{cm}^2)(\text{sec})$.

By representing the thermal flux in both the concrete and steel-water regions with a simple exponential function, the capture gamma fluxes at the boundaries of each region are found by Eqs. (4) or (5) which include linear buildup and assume slab geometry.

$$\Phi\gamma(x=0,t) = \frac{Q_3}{2k} \left\{ e^{kt} E_1(u_s t) - \ln(1-v) - E_1[u_s t(1-v)] \right. \\ \left. + \frac{k}{k-u_s} \left[e^{(k-u_s)t} - 1 \right] \right\} \quad (4)$$

$$\Phi\gamma(x=0,t) = \frac{Q_3}{2k} \left\{ e^{kt} E_1(u_s t) - \ln(v-1) - E_1[-u_s t(v-1)] \right. \\ \left. + \frac{k}{k-u} \left[e^{(k-u_s)t} - 1 \right] \right\} \quad (5)$$

where

$\Phi\gamma(x=0,t)$ = gamma energy flux, Mev/(cm²)(sec)

$v = k/u_s$. Use Eq. (4) when $v < 1$ and Eq. (5) for $v > 1$

k = slope of ϕ_{th} , cm⁻¹. If $\phi_{th} \rightarrow 0$ as $x \rightarrow t$, k is either negative or positive according to whether $x = 0$, or $x = t$

u_s = gamma linear absorption coefficient of the region for E , cm⁻¹

t $\equiv$ region thickness, cm

$Q_3(x=0,t) = \phi_{th}(x=0,t) \sum_c N E$, volumetric gamma source term at either interface of the region, Mev/(cm³)(sec)

$\sum_c$ = thermal neutron capture cross section, cm⁻¹

N = photons/neutron capture at energy, E

E = capture gamma energy, Mev/photon

The selected shield, in the radial direction (Fig. 4) consists of a 30-cm beryllium reflector, a 37-cm water zone, and a 32.4-cm steel-water region followed by 8.5 ft of magnetite concrete (ρ -3.53 gm/cc). The cross sections for these materials are taken from ANL-5800.⁽⁷⁾ The steel-water region which was homogenized for all flux calculations is composed of steel from the internal thermal shield (2 in.), the pressure vessel (1 in. assumed), and the external thermal shield (6 in.) plus 3.75 in. of water. The space between the pressure vessel and the external thermal shield has been neglected. The magnetite concrete is assumed to have only 50% of the normal concentration of water found in this type concrete.

Figures 18, 19, and 20 show the variation of the gamma and neutron fluxes through the radial shield along the horizontal center line of the core. Figure 18 is a plot of the core gammas combined with the thermal shield capture gammas starting from the inner edge of the beryllium and ending at the interface of the thermal shield and the magnetite concrete. From these curves the heat generation rates in the various regions can be obtained. The gamma fluxes at the interface of the beryllium are raised by a factor of 1.5 over the value calculated by Eq. (1) to account for the influence of the core power distribution both radially and axially at the outer surface of the core. The heating rates resulting from the gamma fluxes are listed in Table VII for the beginning and end of each region.

Table VII

GAMMA HEAT GENERATION RATES

<u>Region</u>	<u>Thickness, cm</u>	<u>Inside Surface, watts/cc</u>	<u>Outside Surface, watts/cc</u>
Beryllium	30	30	0.14
Steel (1)	2.54	1.2	0.60
Steel (2)	2.54	0.60	0.315
Steel (3)	2.86	0.315	0.16
Steel (4)	1.27	0.16	0.11
Steel (5)	6.35	0.11	0.025
Steel (6)	6.35	0.025	0.0056
Steel (7)	1.27	0.0056	0.0040
Magnetite Concrete	259	0.0021	

Figure 18 shows the total gamma flux versus the radial distance from the interface of the first thermal shield to the outside of the concrete. The total gamma flux, which is the sum of the core gammas, the thermal shield capture gammas, and the magnetite concrete capture gammas, is $10 \text{ Mev}/(\text{cm}^2)(\text{sec})$ at a concrete thickness of 259 cm (8.5 ft). This conforms to the design criteria specified in Table VI. This magnetite concrete is about 1.5 ft thicker than required for neutron fluxes to meet the criteria stated in Table VI as shown by Fig. 19.

The bulk of the radial shielding is adequate for the design criteria. There are areas of the design, however, that require more detailed investigation. One of these areas is the water gap between the beryllium and the inside thermal shield. The thickness of this gap was chosen on the basis of an initial design configuration. Placing the inside thermal shield too close to the beryllium causes capture gammas of high energy that could control the shielding requirements. The water gap lowers the thermal and fast flux sufficiently so that the controlling factor in heating of the vessel is the gamma radiation from the core. Figure 19 shows that the capture gammas are lower than the core gammas at the interface of the first section of thermal shield. Thus, the inner water gap and the vessel can be made smaller than presently shown.

The internal section of the thermal shield conforms to the pressure vessel contour from the inlet water pipe to a height of 6 ft above the horizontal center line of the core. The external thermal shield is joined to the pressure vessel floor plate and extends vertically to a height of 4 ft 4 in. above the same center line. The magnitude of the permissible gamma fluxes (Table VI) impinging upon the concrete established the internal and external shield heights. Because of the distance involved, the calculations are made using a spherical core of radius 28 cm as a source instead of the solid cylinder core described previously.

b. Above Core

The core gammas on the axial center line above the core are determined by the "exterior on end" equation for a solid cylinder.⁽⁹⁾ The core gammas are attenuated to a value of 10 Mev/(cm²)(sec) by a column of water 15.5 ft in height and a lead plug 12 in. thick.

The N¹⁶ activity at the top of the lead shielding plug is 10% of the core gammas. It is assumed that the water below the steel baffle plate (10.5 ft above the core) contained the same activity as the water leaving the core. This resulted in an infinite plane source of 4.7×10^8 Mev/(cm²)(sec) at the baffle plate. No shielding allowance was made for the steel baffle plate. The activity of the water above the plate is considered to be insignificant.

c. Below Core

The shielding arrangement below the core is similar to the shielding in the radial direction. There is 2 ft of water along the diagonal between the bottom of the core and the point where the internal thermal shield terminates. The remainder of the shield consists of the internal thermal shield, a 3-inch vessel floor plate, a lead slab containing cooling coils equivalent to 1.5 in. of lead and 5 ft of magnetite concrete. The shield within the vertical inlet coolant pipe is formed by a 7 ft water column and a 12-inch lead plug which is extended 6 in. in radius beyond the pipe wall. A 3-inch lead shadow shield is placed below the horizontal inlet coolant pipe to partially compensate for the magnetite concrete displaced by the 24-inch pipe.

The shield below the core reduces the radiation level to 100 mr/hr in the sub-reactor room during operation. At shutdown with a core decay time of 15 min the dose below the lead plug is 1 mr/hr. These values do not include possible streaming around and through the steel control rod drives which penetrate the lead plug.

d. Around Components

The H₂O coolant pipes entering and leaving the reactor must be shielded largely because of the N¹⁶ activity. The activity of the coolant exiting the core is calculated by Eq. (6).

$$N^{16} \left(\frac{\text{dis.}}{\text{cc-sec}} \right) = N F \sigma \left(\frac{1 - e^{-\lambda t}}{1 - e^{-\lambda T}} \right) \quad (6)$$

where

N = number of O^{16} atoms in coolant, atoms/cc

F = average fast neutron flux in the core (fission spectrum),
n/(cm²)(sec)

σ = microscopic cross section for $O^{16}(n,p)N^{16}$ reaction, cm²/atom

t = coolant time in core, sec

T = coolant time, sec

λ = N^{16} decay constant, sec⁻¹

The N^{16} activity at the top of the core is 2.6×10^7 Mev/(cm³)(sec).

The gamma fluxes from the 24-inch inlet and outlet coolant pipes of the reactor are attenuated to 10 Mev/(cm²)(sec) by the utilization of 3.75 ft and 6 ft of magnetite concrete, respectively. Lead or steel shadow shields are used at coolant pipes at certain critical locations.

2. During Shutdown

Calculations of shutdown conditions for fuel are arbitrarily based on a 1000-hour fuel cycle (actual fuel cycle approximately 240 hr) at 100 Mw. Fuel is assumed to be exchanged two hours after shutdown. A water height of 11 ft is required above the fuel as it is placed in the fuel discharge chute. The cumulative dose received by the operator during a complete core unloading is about 4 mr. This calculation is based upon the dose that is received over a period of 20 sec from each of the 48 fuel subassemblies shielded by 11 ft of water.

The core location is 15.5 ft below the water surface and results in a dose of only 0.3 mr for the fuel removal operation. The contribution from the photoneutrons in the beryllium is less than the shutdown radiation from the fuel. Investigations have not been made of probable activity levels from coolant impurities or deposits on the inside of the vessel.

The fuel basket located at the outlet of the fuel discharge chute is adjacent to the canal wall facing the reactor. In order to reduce the dose from a fuel bundle consisting of 6 subassemblies to 0.1 tolerance in the sub-reactor room, the canal between these two points is 7 ft thick.

All of the spent fuel assemblies from a core are grouped together in the canal storage rack containing boron or cadmium to prevent criticality. The shielding calculations for a canal (Fig. 21) are based on the source strength of a complete core having a 3-hour decay after an assumed operating time of 1000 hr. It is not necessary to consider previously removed fuel as a source since its activity is at least a factor of 30 lower.

The radiation at the surface of the water in the canal affects the main floor area where a low background is desired. A water column, 17 ft high, reduces the dose from a freshly discharged core to 0.013 tolerance at the surface of the water. If a height of 3 ft is allowed for fuel height, the floor of the canal is set at 20 ft.

Ordinary concrete, 7.5 ft thick, is used for the lower portion of canal wall to reduce the dose to 0.1 tolerance at the outer wall. The coolant lines are contained in a pipe tunnel along the canal wall (Fig. 21). The water flowing through the lower coolant pipe leading to the heat exchanger building is assumed to have the same N^{16} activity as the water leaving the core. An ordinary concrete wall, 6 ft thick, is required to reduce the radiation from this 24-inch outlet pipe to 0.1 tolerance in the active storage and service area in the basement level. Because of distance attenuation, a ceiling only 6 ft thick is needed above the pipe for a dose which is 0.01 tolerance at the main floor level; however, an additional foot has been added to reduce radiation streaming through the water zone of the canal.

IV. HEAT REMOVAL - FLUID FLOW

A. Steady-State Operation

1. Core Heat Analysis

The fission heat generated in the core (100 Mw) is assumed to be distributed in the same manner as the thermal flux in the core, since nearly all of the fissions are caused by the thermal neutrons. However, the flux plots shown (Figs. 14 and 15) are not representative of the power distribution. The total maximum to average flux ratio is derived from the problem data (PDQ 4809) which is the worst distortion case. A "chopped" cosine distribution is assumed axially for calculation of temperature distributions throughout the flow channels. These surface and bulk water temperatures are shown in Fig. 22. The heating effects due to flux peaking at the channel ends are not shown in Fig. 22. However, a calculation showed that the surface temperature at the end of the hottest channel did not exceed that of the maximum hot spot downstream.

The core flow of 13,000 gpm is upward through the fuel assemblies at a velocity of 35 ft/sec. The high average velocity and the narrow flow channel (0.057 in.) create a high convection coefficient, but the narrow channels demand accurate fabrication tolerances and allowable plate deflection must be small. Hydraulic testing of 0.050-inch plates with 0.110-inch spacings has shown that such arrays satisfactorily withstand coolant velocities up to 49 ft/sec.⁽¹⁰⁾

A considerable safety margin exists between core outlet pressure and saturation pressure at the maximum hot channel surface temperature for prevention of boiling. Should it develop that the velocity in the channels is too high or the channels too narrow, one or both may be changed leaving the core inlet temperature constant and allowing the aluminum surface temperature to increase to as high as 400°F. The core inlet temperature is fairly high (135°F), for a research reactor, in order to minimize heat exchanger cost.

Core heat transfer and fluid flow characteristics are shown in Table VIII.

2. Flow Balance in Reactor Vessel

The arrangement of beryllium reflector, H₂O, and thermal shields within the reactor vessel using a single-pass flow requires the coolant velocities in the various channels be properly proportioned. The total primary flow is 15,000 gpm, of which 2,000 gpm bypasses the core and flows through three general regions: the beryllium reflector, the water annulus between the reflector and the thermal shield, and the internal thermal shield sections. Table IX lists the flow distribution with the flow orificing located in the upper grid structure.

Table VIII

HEAT TRANSFER AND FLUID FLOW

Power, Mw	100
Total primary flow, gpm	15,000
Core flow, gpm	13,000
Average H ₂ O core velocity, ft/sec	35
Inlet temperature, °F	135
Outlet temperature, °F	188
Core pressure drop, psi	37.3
Core outlet pressure, psia	200
Average power density, Mw/liter	1.13
Maximum power density, Mw/liter	1.90
Average heat flux, Btu/(hr)(ft ²)	580,000
Maximum heat flux, Btu/(hr)(ft ²)	980,000
Axial maximum/average flux	1.29
Radial maximum/average flux	1.30
Hot channel factors	
bulk rise	1.29
film rise	1.56
Maximum surface temperature (hot channel) °F	315
Maximum fuel temperature (hot channel) °F	365

Table IX

PRIMARY COOLANT FLOW DISTRIBUTION
IN REACTOR VESSEL

(in gpm)

Core	13,000
Beryllium reflector	800
Water annulus	700
Thermal shield channels	500
	<hr/>
Total	15,000

B. Unsteady-State Operation

Normal reactor shutdown is accomplished without reduction of the main coolant flow until the reactor power has decreased to about 5 Mw. At this time, one main coolant pump is shut down and the other is throttled to a value comparable to the rating of the shutdown pump which provides 1500 gpm for removal of decay heat from the core. The shutdown pump operates normally on the main plant power system. The same arrangement and procedure applies to the secondary coolant system.

The loss of primary or secondary flow by reason of a main power failure or failure of the main coolant pumps requires that some emergency measure provide coolant to the reactor core for prevention of structural damage due to overheating. Power failure causes a reactor scram, and emergency power is supplied by a diesel-generator to the shutdown pumps. The time between the loss of power and the provision of adequate flow to the core is determined by the "coast down" rate of the main primary flow (Fig. 23). This curve was derived from consideration of the rate of decrease in momentum of the primary coolant. The required primary flow is based upon a maximum allowable surface temperature of the fuel elements of 190° F. Assuming that system pressure is also lost, this allows emergency cooling without boiling in the core. From these curves, it is seen that approximately 20 seconds is allowed for diesel-generator start-up.

In the event of a rupture in the primary system external to the reactor vessel, the water in the vessel up to the level of the main outlet line is retained because of the loop made by the inlet pipe. A small circulation and vent line from the top of the loop to the upper portion of the reactor vessel prevents the water from siphoning to lower levels of the primary loop. The only means of cooling the core is natural convection. Since the required orifices at the outlet end of the reflectors and thermal shields offer a high resistance to flow, another path is provided for completion of a natural convection loop. This path is a pipe which leads from the pressure vessel wall above the core to the core inlet header. A check valve in this line prevents flow by bypassing the core under normal conditions.

An emergency cooler-condenser in the reactor shield is included in this bypass line and provides some heat removal capacity by virtue of secondary flow from the process water storage tank (Fig. 9).

C. Cooling of Spent Fuel

After reactor shutdown the spent fuel elements are removed from the core and transferred to the canal for storage. Figure 24 indicates that the total fission product decay heat output 1.4 hours after

shutdown is 0.8 Mev. At this power level, the bundle of fuel elements can be cooled by natural convection while above the core, in the fuel discharge tube, and in the canal. With a water temperature of 100°F, the maximum surface temperature of the fuel plates is about 205°F.

Two valves at the bottom of the discharge tube function as a pressure lock to permit transfer of spent fuel from the reactor vessel to the canal. This pressure lock must be connected to a pressure relief mechanism to accommodate bulk expansion of the water while a fuel bundle is in this section. A cooling line and heat exchanger are provided to remove heat from the lock section should a fuel bundle become trapped for a period longer than one or two minutes.

D. Stress Analysis

1. Pressure Vessel

The design of the vessel conforms to the ASME Unfired Pressure Vessel Code.⁽¹⁾ The design pressure was set at 300 psi based on a 20% over-pressure allowance for the core inlet pressure. The thermal stress is calculated based on the heating rates determined in Section IV B. The heating rate used for the vessel is

$$\ddot{q} = 0.47 e^{-0.26x}$$

where,

$\ddot{q}$ = volumetric heat generation rate, watts/cc

x = vessel wall thickness, cm.

It is estimated that about 3500 psi tensile stress is sufficient allowance for the beam holes and similar attachments to the vessel.

The combined pressure and thermal stress is a maximum at the inside of the vessel since it is considered to be insulated on the outside. The total stress is given by:⁽¹¹⁾

$$\sigma_c = \frac{(PR_0/12t) - 0.4P}{E} + \frac{E_y \alpha \Delta T'}{1-u}$$

$$\nabla T' = \frac{q_0}{\Sigma K} \left[\frac{e^{-\Sigma t} - 1}{\Sigma^2 t} + \frac{1}{\Sigma} - \frac{te^{-\Sigma t}}{2} \right]$$

where

σ_c = combined stress at inside of vessel

P = design pressure, psi

R_0 = outer radius of vessel, ft

t = thickness of vessel wall, ft

E = joint efficiency

E_y = modulus of elasticity, psi

α = linear coefficient of expansion, in/(in.)(° F)

u = Poisson's ratio

q_0 = volumetric heat generation rate at inside surface of vessel
Btu/(hr)(ft³)

Σ = slope of heating curve, ft⁻¹

k = thermal conductivity of vessel wall, Btu/(hr)(ft)(° F)

$\Delta T'$ = average temperature difference between inside surface and
average temperature of wall, ° F

A wall thickness of 1.125 in. results in a minimum combined stress. The contributing stresses are summarized in Table X.

Table X

STRESSES IN PRESSURE VESSEL WALL

(in psi)

Pressure stress	11,756
Thermal stress	3,314
Combined stress	15,070
Allowable stress	18,750
Beam hole allowance	3,680

2. Center Core Box

The center box structure (Fig. 5) serves to position the fuel subassemblies in the core. The box is aluminum, 4.75 in.² by 20.75 in. long, with a wall thickness of 0.207 in. The sides of the structure

must also be capable of withstanding the thermal stress caused by a heating rate of 166 watts/cc and the pressure forces caused by the orificing of flow through the center of the box. Orifices are incorporated at both ends of the box-like structure to prevent the full pressure differential of 37 psi from acting on the ends of the structure.

The maximum stress in the center box (~ 4000 psi) is caused mainly by the pressure forces. The heating rate results in a thermal stress less than 230 psi. The structure is capable of withstanding a much higher pressure force. The center box is reinforced vertically opposite the locating grooves to prevent weakening by the grooves.

3. Beryllium Reflector

The heating rate in the beryllium varies from 30 watts/cm³ at the inside face to 0.14 watts/cm³ at the outside surface.

The thermal stress conditions existing during operation must be examined by a more accurate analysis of the thermal gradients caused by the many holes in the beryllium. The influence of the rate of start-up and shut-down on the beryllium has not been investigated; however, it is recognized that the normal operational procedure of the reactor may be dependent upon the thermal stress in the beryllium.

V. COST ESTIMATE

A. Capital Cost

The capital cost estimate of the reactor facility is itemized in Table XI.

Table XI

REACTOR COST ESTIMATE

	<u>Sub-Total</u>	<u>Total</u>
<u>BUILDING COST</u>		
Construction Cost		
Improvement to Land	\$ 66,000	
Cost of Building		
Reactor (includes external utilities)	1,666,000	
Equipment and Heat Exchanger		
Building	856,000	
Office and Administration Building	448,000	
Other Structures (second cooling system, stack and stack building)	712,000	
Outside Utilities	405,000	
Equipment	107,000	
Total "A"	4,260,000	
A-E Cost and ANL Liaison (10% of Total "A")	426,000	
Indirect Cost (1% of Total "A")	44,000	
Total "B"	4,730,000	
Contingencies (15% of Total "B")	710,000	
Building Cost		\$ 5,440,000
<u>REACTOR COST</u>		
Pressure Vessel Assembly	\$ 346,000	
Reflector	600,000	
Control Systems	246,000	
Shielding	274,000	
Instrumentation	865,000	
Fuel and Control Rod Removal Devices	57,000	
Experimental Facilities and Equipment	820,000	
Primary Cooling System	926,000	
Hot Liquid Waste System	86,000	
Canal and Transfer Coffin	240,000	
Acceptance Test and Replacement	346,000	
Miscellaneous	100,000	
Health Physics and Area Monitoring	64,000	
Total "C"	4,970,000	
A-E Cost and ANL Liaison (12.5% of Total "C")	630,000	
Total "D"	5,600,000	
Contingency (25% of Total "D")	1,400,000	
Reactor Cost		\$ 7,000,000
Project Cost 10%		12,440,000
Escalation		1,260,000
Grand Total		\$ 13,700,000

B. Operating Costs

The operating costs exclusive of fuel, are presented in Table XII. The wages and salaries are based on 75 persons with an average annual allowance per person of \$12,000. This includes indirect charges. The power cost is based on an operating electrical consumption of 2000 kw at 8 mil/kw-hr and 300 operating days per year. The water cost is based on a consumption rate of 1200 gpm, a unit cost of \$0.10/1000 gal and 300 operating days/year.

Table XII

ANNUAL OPERATING COST

Wages and Salaries	\$ 900,000
Power	110,000
Water	50,000
Misc. Operating Supplies	100,000
	<hr/>
	\$1,160,000

The annual fuel quantities (300 operating days) are estimated as follows:

kg of U^{235} as UF_6 , including 10% scrap recycle allowance:	165
Number of fuel subassemblies	1320
Burnup, based on 1.25 gm/Mwd:	37.5
kg of U^{235} recovered from spent fuel:	111
kg of U^{235} lost:	1.5

Based on unit charges of \$100/kg of U^{235} to convert from UF_6 to U-Al alloy, \$200/subassembly to fabricate, \$1000/kg of U^{235} to recover scrap, and \$4000/kg of U^{235} to process spent fuel, the annual fuel cost is of the order of \$600,000, exclusive of a burnup charge.

The total inventory of U^{235} in all stages, in fabrication, in reactor, in decay cooling, and in process is estimated at ~90 kg.

VI. FUTURE STUDIES

The design presented requires no major development phases of a basic materials or design nature prior to the start of a final system design. However certain basic development efforts would prove advantageous.

1. The fuel shown in this report is a plate element-box type sub-assembly. This is the most highly developed type for high power density, highly enriched research reactor cores, and this type of fuel can be used successfully. However, the reactor operation and design can profit by developments along two lines, (a) fuel elements lending themselves to variable concentration of fuel for flux flattening purposes, and (b) fuel subassemblies better adapted to the circular geometry of the annular cores. Such development efforts do not require new materials; the use of aluminum clad in combination with aluminum-uranium alloy or aluminum-uranium oxide dispersions is adequate. It should be noted that the reactor design presented features a replaceable internal structure in order that future improved fuels may be accommodated.

2. The reactor design has a heavily loaded core; future studies may well increase the loading. The effect of beam holes may prove to be a very sensitive function of position and difficult to estimate. Consequently, a critical experiment is considered a worthwhile investment.

The results obtained from either of these two experimental efforts would benefit reactors other than the proposed design. The fuel developments can be applicable to many reactors. The critical studies most certainly would be of great benefit for the annular core reactors which are being given special consideration for both research and test reactors.

Of the more prosaic efforts remaining, the following are some of the more important:

Design

1. Detailed study of the reactor vessel with its internal structure and means of support including a stress analysis of the more critical parts.

2. The design and operating characteristics of the external reflectors, particularly if they are solid materials, i.e., graphite or beryllium.

3. Development of experimental facilities and means of handling experiments. The major effort in this area is the central thimble and its effect on the reactor control system. This facility probably requires some degree of mock-up.

4. The entire system of fuel handling.
5. Further efforts on the requirements for the sizes of and the type of construction of the various buildings. The general problem of materials handling needs study.
6. The design of the coolant systems with emphasis on control and a more optimum balance between the primary heat exchanger and cooling tower operating condition.
7. More details on possible emergency cooling situations and means of handling them.
8. Obtain more information about materials, i.e., radiation damage effects and effects of materials (corrosion) on water quality control.

Physics

1. Establish critical properties of a final design, including the effects of core and neutron temperature, etc.
2. Establish the effects of perturbations, such as control rod movement, voids in H_2O , insertion of samples for irradiation, beam holes and changes in temperature and neutron lifetime.
3. Make a detailed study of the fuel burnup problem and determine an initial loading. In this connection it may prove profitable to include a study of cores with an outer reflector of H_2O , since these cores, with a higher clean, cold critical mass, have less sensitivity to an increase in the initial loading.
4. Study possible long range kinetic effects for both temporal and spatial xenon oscillations.
5. Conduct detailed shielding studies, particularly in areas where streaming may exist or where the normal shield may require reinforcement with shadow shielding. Perform similar studies on activation of the primary water system and the shielding required at all parts.

Heat Transfer and Fluid Flow

1. Re-evaluate the heat generation throughout the reactor vessel under conditions of different phases of core cycle, with particular emphasis on the influence of experiments and control rod positions on the heat generation patterns with the core.
2. Study and make hydrodynamic tests of the final fuel designs.

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FIG. 1
FUEL ASSEMBLY

FIG 3
PRESSURE VESSEL AND
INTERNAL STRUCTURE

FIG 5

SECTION AT
HORIZONTAL ϵ

CANAL

NOTE:
Sample Removal Tool Similar to
That Shown on Fig 6

Fig 7
Typical Vertical
Irradiation Facilities

FCV - FLOW CONTROL VALVE
 PCV - PRESSURE CONTROL VALVE
 LC - LEVEL CONTROL

FIG. 9
 COOLANT SYSTEM
 FLOW DIAGRAM

FIG. 11
REACTOR BLDG
BASEMENT - PLAN

SECTION A-A
(SEE FIG 10 II)

FIG 12
REACTOR BLDG
ELEVATION

FIG. 13
PLOT PLAN

FIG. 14
RADIAL THERMAL FLUX DISTRIBUTION AT AXIAL CENTER.
INITIAL LOADING: 4.3 kg U²³⁵. BURNABLE POISONS. NO BURNUP.

FIG. 15
 AXIAL THERMAL FLUX DISTRIBUTION ABOVE REACTOR MIDPLANE AT $R = 15$ cm.
 INITIAL LOADING: 4.3 kg U^{235} . BURNABLE POISONS. NO BURNUP.

FIG. 16
RADIAL THERMAL FLUX DISTRIBUTION AT REACTOR MIDPLANE
INITIAL LOADING: 4.3 kg U^{235} . BURNUP: 1 kg U^{235} .

FIG. 17
AXIAL THERMAL FLUX DISTRIBUTION ABOVE REACTOR MIDPLANE AT $R = 15$ cm.
INITIAL LOADING: 4.3 kg U^{235} . BURNUP: 1 kg U^{235} .

FIG. 18
RADIAL GAMMA FLUX ENERGY DISTRIBUTION AT REACTOR MIDPLANE

FIG. 19
RADIAL GAMMA FLUX DISTRIBUTION AT REACTOR MIDPLANE

FIG. 20

RADIAL DISTRIBUTION OF NEUTRON FLUX AT REACTOR MIDPLANE

NOTE: ALL DIMENSIONS IN FEET

FIG. 21
FUEL CANAL SHIELDING ARRANGEMENT

FIG. 22
AXIAL TEMPERATURE DISTRIBUTION IN CORE AT 100 Mw

FIG. 23
AVAILABLE AND REQUIRED PRIMARY COOLANT FLOW FOLLOWING LOSS OF PRIMARY PUMP POWER

FIG. 24
FISSION PRODUCT DECAY HEAT GENERATED IN CORE AFTER OPERATION AT 100 Mw FOR 1000 hr