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UNITED STATES ATOMIC ENERGY COMMISSION

A LOW COST PHYSICS AND ENGINEERING
TRAINING REACTOR

Reactor Design and Feasibility Study

By

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August 1956

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OAK RIDGE SCHOOL OF REACTOR TECHNOLOGY

Reactor Design and Feasibility Study

A LOW COST PHYSICS AND ENGINEERING TRAINING REACTOR

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August 1956

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PREFACE

In September, 1955, a group of men experienced in various scientific and engineering fields embarked on the twelve months of study which culminated in this report. For none of those months, formal classroom and student laboratory work occupied their time. At the end of that period, these eight students were presented with a problem in reactor design. They studied it for ten weeks, the final period of the school term.

This is a summary report of their effort. It must be realized that, in so short a time, a study of this scope can not be guaranteed complete or free of error. This "thesis" is not offered as a polished engineering report, but rather as a record of the work done by the group under the leadership of the group leader. It is issued for use by those persons competent to assess the uncertainties inherent in the results obtained in terms of the preciseness of the technical data and analytical methods employed in the study. In the opinion of the students and faculty of ORSORT, the problem has served the pedagogical purpose for which it was intended.

The faculty joins the authors in an expression of appreciation for the generous assistance which various members of the Oak Ridge National Laboratory gave. In particular, the guidance of the group consultant, L. G. Alexander, is gratefully acknowledged.

Lewis Nelson

for

The Faculty of ORSORT

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ABSTRACT

This report covers the conceptual design of a low cost training reactor for the instruction of physicists and engineers. It is conceived as an instructional tool for a course such as that given at the Oak Ridge School of Reactor Technology. As envisioned in this report, the reactor is of a modified pool type. It is designed for a maximum power level of one megawatt. This arrangement will accommodate engineering experiments, shielding experiments, and critical experiments as well as being useful as a neutron and gamma source.

ACKNOWLEDGEMENTS

The authors of this report would like to express their thanks for the assistance of many members of the staff of ORNL. Particular gratitude is due L. G. Alexander for his patient and competent guidance as advisor to the group. Special thanks are also due Lewis Nelson, Herbert Pomerance, Keith Henry, Frank Buford, and Elizabeth Johnson for their assistance on particular aspects of our problem.

Chapter I

1.0 Introduction

When the requirements of a nuclear training program are examined, it is immediately apparent that none of the existing reactor configurations are entirely satisfactory. The design envisioned in this report is an attempt to satisfy the many conflicting demands placed on a reactor under such a program as that of ORSORT. While the primary consideration has been the requirements of ORSORT, the attempt has been made to keep those of a university in mind also.

In order to keep the cost of the reactor within reason and still meet these conflicting demands, a large number of compromises have been made. The basic advantage of the pool type (the ability to handle "hot" components under water) has been retained. At the same time the top structure, which is in the way in so many cases, has been eliminated. The inclusion of a thermal column and beam ports in the wall at the end of the pool provides for many types of experiments which cannot conveniently be performed in the pool.

The arrangement chosen lends itself to engineering experiments requiring the generation of a appreciable amount of power as well as the usual types of physics and shielding experiments now requiring separate facilities at ORNL. These objectives have been met at some cost in refinement for some purposes, but it is hoped that the penalties accepted will not prove excessive.

Chapter 2

2.0 Engineering Description

2.1 Requirements

The problem envisioned in the design of a reactor suitable for use by colleges and universities in conjunction with courses in nuclear engineering and also useful as a research tool in physics, chemistry, biology, etc. It is desirable that the reactor cost between \$100,000 and \$250,000, exclusive of the building and experimental equipment. Because of its use as an instructional tool, it must necessarily be a low hazard machine. Because of the fact that the reactor is to be used as a training device for engineers, it should possess the characteristics of a power reactor. This requires that the power output be appreciable. For the purposes of this design the power output has been set as one megawatt maximum. It should be arranged so that a large variety of experiments, other than engineering experiments, can be performed, i.e., shielding experiments, critical experiments, and usefulness as a neutron and gamma ray source.

2.2 General Description

As will be seen by reference to the drawing showing the general arrangement of the reactor (Figure 2.2), the Physics and Engineering Training Reactor (PETR) bears a generic relation to the ORR and also to the Bulk Shielding Facility. The reactor core is in a fixed position in the bottom of a water-filled pool. In this arrangement the control rods enter from the bottom of the structure; the rod drives are located in a sub-reactor room which also contains various auxiliary equipment besides the hydraulic rod drives. The wall adjacent to the core is equipped with a thermal column and also with two beams ports to

Fig 2.1
P.E.T.R.
56

allow for physics experiments which are better run in the "dry" rather than submerged in the pool as is necessary with the usual "swimming pool" type structure. The elimination of the top hamper associated with the normal pool arrangement allows for much easier insertion into the core of various experiments, such as test loops, which is not feasible if the control rod drives are arranged above the core.

2.3 Fuel Elements

The fuel elements for the PETR are identical to those of the ORR with the exception of the top end box. These elements contain 140 gms of U-235. The elements have 19 fuel plates 0.050 inches thick (.020 inches U-Al alloy and .015 inches cladding of Al) having an active length 24 inches by 2-3/4 inches wide. In the case of the top end box, they have been modified to the form used in the BSR. This allows the elements to be handled in the same way - by hand tools. The arrangement of the fuel elements is shown in Drawing 2.3.

2.4 Core Support Structure

The core support structure is shown in Figure 2.4. In this structure the fuel elements are supported by a replaceable grid plate which is attached to the structure proper by bolts. This feature allows the arrangement of the fuel element grid to be changed with facility. Underneath the grid plate is a pressure-tight structure with a large gate valve to allow access of pool water to the underside of the grid plate. This arrangement allows the core to be cooled either by natural convection (with the gate valve open) or by forced circulation when the valve is closed. It also makes possible the operation of the reactor as a pressurized water reactor by enclosing the top of the core within a pressure shell. It will be noted that this core structure

MATERIALS LIST

ITEM NO.	DWG. NO.	QTY.	NAME	SIZE	MATERIAL
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FIG. 2.3

REACTOR ASSEMBLY & SECTIONS	D-26043
REFERENCE DRAWINGS	DWG. NO.

PHYSICS ENGINEERING TRAINING REACTOR
PETR
FUEL ELEMENT DETAILS

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also carries the control rods. The arrangement which we have shown allows the entire operating reactor core structure to be shop fabricated as one assembly so that there is no question of proper alignment of the various components. No field work is necessary to insure the proper operation of the mechanical parts of the machine. The entire structure, wherever possible, is to be fabricated of aluminum in order to avoid problems of corrosion and activation. The structure is designed for an operating pressure of 100 psi at 300°F.

2.5 Control Rods and Drive

The control rods for PETR will be essentially the same as those used in the BSR. They are driven by oil hydraulic cylinders on the bottom of the core structure and extend downward into the sub-reactor room. The rods, when withdrawn, extend above the top of the core and, when fully inserted, are flush with the top of the active length of the fuel elements. This arrangement achieves fail-safe operation since gravity, rising pressure, and assist springs, on loss of power all act to insert the control rods. The shaft, which is integrally fastened to the control rods, extend through the hollow piston rod of the hydraulic drive cylinder and is attached at its lower end. In order to change a control rod, it is necessary to remove only the nut at the bottom of the piston rod. This frees the control rod so that it may be lifted out through the top of the core structure. The leakage of water into the sub-reactor room is prevented by a cap which covers the end of the rod after the control rod is disengaged.

The scram insertion of the rods is accomplished by a separate hydraulic circuit which is pressured from an accumulator which operates on the upper side of the actuating piston. The scram action is also assisted by compression

springs which are arranged above the cylinder. The position of the control rods is remotely indicated at the control bench by the means of a selsyn which is driven by the bottom end of the piston rod. In order to avoid the accumulation of radioactive material in the assist spring pockets, a bleed line is attached to the spring pockets just above the sealing gland and led to the demineralizer circuit. Although only three control rods are normally required for the operation of the machine, additional rod support structures are built into the core support to allow for other arrangements, if required. A fourth rod drive is provided to be used in the operation of a loop or as a spare.

2.6 Pool

The pool containing the reactor structure is approximately 20 ft. long by 16 ft. wide by 25 ft. deep. The reactor, which is located at one end is viewed through the adjacent wall by two beams ports and a thermal column. The walls containing the beam ports and thermal column are of high density concrete for shielding purposes, as is the slab over the sub-reactor room. At the opposite end of the pool a 16 ft. deep well is provided for the storage of hot elements and equipment when the pool is drained for access. The top of the pool is to be covered entirely with a removable grating floor with the exception of an opening approximately 5 ft. wide from end to end. This arrangement allows all operations to be performed by hand without the use of a crane. The beams which support the grating adjacent to this opening are arranged to receive an instrument carriage which can traverse the length of the pool supporting instruments hung on the centerline of the reactor. The interior of the pool is to be painted with "Amercoat".

2.7 Buildings

Tentatively, the building is conceived to be a steel frame structure approximately 40 ft. wide having 4 bays each 16 ft. long. The interior would be sheathed with Robertson "Q" panels. An adjacent structure approximately 16 X 34 having two floors would be built of concrete block construction. This structure would contain the reactor controls and auxiliary equipment such as motor control centers, etc. It would also house lavatories, office space, and maintenance shop. The operating control panel is to be mounted on a balcony of the second floor of the structure adjacent to the pool and at the centerline of the core. The controls for the rod drives are located on a low bench in front of the operator with the indicating instruments on the side (see Fig. 2.7). This enables the operator to see the entire operation, and at the same time removes the controls from the floor immediately adjacent to the pool where they would interfere with access for operation. The building is to be provided with a monorail type electric hoist on a transverse bridge. This allows the crane to service the entire operating area of the building. The hoist would be of approximately 2-ton capacity. The experimental area associated with the beam ports and thermal column will be open at the top to allow hoist access to this lower floor. Access to the sub-reactor room is through a hatchway in the floor of the experimental area.

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ITEM NO.	QTY.	NAME	SIZE	MATERIAL

FIG. 2.7	
REACTOR ASSEMBLY & SECTIONS	D-26045
REFERENCE DRAWINGS	DWG. NO.
PHYSICS ENGINEERING TRAINING REACTOR	
PETR	
BUILDING STRUCTURE	
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Chapter 3

3.0 Cooling Water System

3.1 Description of Cooling Water System

The cooling water system for the PETR consists of two separate loops. The primary loop is for operation as a pressurized water reactor. The water is withdrawn from the bottom of the grid plate through a centrifugal pump from whence it goes to the tube side of a shell and tube heat exchanger and is returned to the top of the core. The secondary loop takes water from the pool through a centrifugal pump, through the shell of the heat exchanger, to a fin-fan cooling tower and returns it to the pool. For unpressurized operation, the system is so manifolded that it can be operated either with natural circulation or with forced circulation through the core. With natural circulation, water is removed from the pool through a skimmer, directed to the fin-fan cooling tower, and returned to the pool through a series of diffusing jets arranged to direct the water across the top of the core to achieve maximum mixing of the hot water rising from the core with that of the pool. The system can also be arranged so that the pool may be cooled through the heat exchanger with the fin-fan unit operating with the shell of the heat exchanger as a closed loop. Forced circulation cooling may be achieved in the unpressurized condition by closing the 12 inch gate valve at the bottom of the reactor structure and drawing pool water down through the core from whence the water could be passed either to the heat exchanger and indirectly cooled by the fin-fan unit or directly to the cooling tower and hence to the pool. The flow of water in the secondary system, which normally removes water from the pool through the skimmer, is directed through a bag-type filter in order to remove foreign material which might become activated if it were

allowed to accumulate in the pool. A demineralizer is provided which operates on a by-pass arranged in such a way that the demineralizer can be hooked into either the primary or the secondary cooling system. Under pressurized operation, the demineralizer would be in the primary system. Under non-pressurized operation, the demineralizer would be in the secondary system. An elementary schematic diagram of the cooling water system is shown in Figure 3.1. The details of the arrangement and the detailed calculations for the cooling system are covered in Appendix A.

3.2 Natural Convection Cooling

Cooling of the core by natural convection was calculated for the case of a 5 x 5 array of fuel elements operating at one megawatt. It was assumed that three of the elements contain rods. It was also assumed that the central element operates at twice the average operating power and that the heat is introduced into the cooling stream as a linear function of distance along the fuel plate with the entering pool water at 100°F. It was found that one megawatt operation would give a velocity of 0.4 feet per second in the fuel section of the element and an 82°F temperature rise across the core. It was determined also that the PETR could be operated at 1.25 Mw with 100 degree entrance water without boiling in the central element of the core.

3.3 Forced Convection Cooling

When the PETR is operated under conditions of forced convection with the system unpressurized, the pumping system will put through the core approximately 31.6 lbs. per second of cooling water. With this flow the temperature rise through the core amounts to 30°F. The cooling tower is sized to dissipate one megawatt of heat with a 30 degree temperature drop through the unit.

ORNL LR Dwg. 15295

ELEMENTARY SCHEMATIC DIAGRAM
OF COOLING SYSTEM

FIG 3.1

Therefore, if the equilibrium temperature of the cool water is 100°F, the water discharged from the core will be at 130°F.

3.4 Pressurized Water Operation

When the PETR is operated as a pressurized water reactor, cooling is by means of the heat exchanger between the core and the cooling tower. Both the primary and the secondary loops are designed to have a 30° temperature difference and a flow of 31.6 lbs. per second. The system is designed to operate at a maximum of 100 psi and 300°F. One Mw can be dissipated, however, with a maximum temperature in the primary loop as low as 170°F and an inlet temperature to the reactor of 140°F. Instrumentation is provided in the piping system to allow flow rates and temperature differences to be measured in order that the power output of the reactor can be determined.

3.5 Demineralizer System

Pure water is important in this system to minimize aluminum corrosion, reduce water activity and minimize impurities which reduce reactor reactivity. Mixed bed deionization gives more and purer water at a lower cost than either conventional deionization or distillation. This type of system has been operating satisfactorily at the BSF for a considerable period of time and has proved to be satisfactory. Purity of the pool water will be maintained by a 6-GPM continuous by-pass of the filter effluent through the ion exchange bed. A demineralizer similar to the Elgin "Ultra Deionizer" will keep the water resistivity to at least one mega-cm. A small filter after the bed is provided to remove any resin carry-over from the deionizer.

Chapter 4

4.0 Core Studies - Physics

4.1 Criticality

Critical masses for five different core configurations of the PETR have been calculated to provide estimates of the fuel required for the reactor. The critical mass ranged from 2270 grams of Uranium 235 for the close-packed arrangement to 4150 grams for the case where a large solid fuel element test loop is inserted in the central region of the core.

The minimum critical mass of the PETR is considerably less than that normally envisioned for a pool type reactor (3500 grams) for the following reasons:

1. The fuel elements were placed closer to one another, thus increasing the fuel/water ratio.
2. The number of fuel plates per fuel element was increases from 18 to 19 and the thickness of the fuel plate cladding was reduced. These changes reduce the volume ratio of aluminum to water in the core.
3. The critical calculations for the PETR did not account for the water channels required in an operating reactor for control rod insertion. The average loss in reactivity due to each of these ports was estimated to be $0.5\% \frac{\Delta k}{k}$.

4.2 Control Rods

Two types of absorber rods control the PETR. A hollow stainless steel sleeve constitutes the regulating rod and provides fine control of the reactor. The average reactivity worth of this rod is $-0.7\% \frac{\Delta k}{k}$.

The safety rods each contain 347 grams of natural boron and perform the dual job of controlling the excess reactivity built into the reactor and "shimming" to compensate for poison build up, fuel burn out, or other reactivity losses. Boron was chosen for the absorber material because of its relatively large absorption cross section in the epithermal range. The average reactivity worth of each safety rod is $-3.5\% \frac{\Delta k}{k}$.

4.3 Operational Loading

The excess reactivity requirements of the reactor are estimated not to exceed 3.5 to 4.0% $\frac{\Delta k}{k}$. This estimate is based on a maximum power of one megawatt and an average operation of four hours per day for a one year period.

Two safety rods and one regulating rod have sufficient control for the above excess reactivity. For extraordinarily large experimental losses, a third safety rod can be added to control the additional reactivity requirements.

Chapter 5

5.0 Biological Shielding

5.1 Introduction

The shielding of the proposed PETR has been designed to attenuate the high intensity neutron and gamma fluxes arising from the core region at one megawatt operation.

In general the shielding is so designed that the dose rate is well below biological tolerance in those areas which are accessible to personnel during full power operation. The biological tolerance dose rate is taken to be 7.5 milli-rem per hour. Based on a 40 hour work week this dose rate gives 300 mrem per week, "the maximum permissible exposure", as recommended by the International Commission on Radiological Protection. This maximum permissible exposure is based on whole body dose from external sources of radiation.

5.2 Physical Layout

The side-hill location for the pool structure as proposed allows for essentially infinite shielding of the core on 3 sides. On the fourth side the gamma and neutron fluxes are reduced to one tenth tolerance by approximately $1\frac{1}{2}$ feet of water and $7\frac{1}{2}$ feet of barytes concrete. The thick concrete wall shall be referred to as the "main shield", and the remaining portion of the pool as the "pool section". (See Figure 2.7)

The pool section is 22 feet long, 16 feet wide, and 27 feet deep. This depth allows for 22 feet of water over the reactor core and gives sufficient attenuation of the direct penetrating gamma radiation in the vertical direction. The fast and slow neutrons will be attenuated even more.

5.3 Dose Rates

The dose rate at the surface of the water over the core is strongly dependent upon the flow pattern of water from the core during full power generation. The flow pattern as described in Appendix A.1.6 will allow the 7-second N^{16} activity induced in the water by the $O^{16}(n,p)N^{16}$ reaction to decay sufficiently before reaching the surface. If the naturally convected coolant water from the core is allowed to rise directly to the pool surface the 6 Mev gamma from this reaction will predominate over the direct gamma radiation from the reactor.

At the end of the pool opposite the main shield a trench is provided for storage of activated fuel elements. The trench provides 13 feet of water over the stored elements when the main pool section is drained.

Directly below the core is a subreactor room which is not accessible to personnel during operation of the reactor. (See Fig. 2.7) Three feet of water and four feet of barytes concrete serves to attenuate the fission product gamma activity from the core so that personnel may enter the sub-reactor room $\frac{1}{2}$ hour after shutdown from extended high power operation.

The main shield is constructed of barytes concrete rather than ordinary concrete so that the thickness necessary to produce the desired attenuation will not become excessive. It is $7\frac{1}{2}$ feet thick at the elevation of the core and tapers to $1\frac{1}{2}$ feet at the top. Direct gamma penetration through the main shield in any direction will experience an equivalent amount of attenuation from water plus concrete. Neutrons from the core will experience an even greater reduction.

Chapter 6

6.0 Experimental Facilities

6.1 General

The experimental facilities described below (a graphite thermal column and two beam ports) will only be used when the core is operating with the pressure vessel removed. It is intended that the radiation level outside the main shield be low enough to allow operation of sensitive research instruments with little additional shielding.

6.2 Thermal Column

The permanent section of the thermal column is 4 by 4 feet in cross section and extends the full $7\frac{1}{2}$ feet thickness of the main shield. To use the thermal column as a research facility the permanent section is extended to the core face by a portable slab of canned graphite interposed between the core and the water barrier. The aluminum water barrier is placed flush with the inner surface of the main concrete shield. Directly behind the water barrier and in contact with it is a 4 inch thick lead curtain. The remainder of the $7\frac{1}{2}$ feet is graphite which is shielded from the concrete by a boral liner followed by a lead liner. Cooling pads are provided in the concrete near the lead liners. The outer end of the thermal column is shielded by a 5 feet square, 10 inch thick lead door lined on the core side with a $\frac{3}{4}$ inch thick sheet of boral. With the door in place the gamma and neutron fluxes are reduced to 1/10 tolerance at the outside surface. Access to the thermal column can be gained either through a small opening provided in the door or by sliding the door completely out of the way. (See Fig. 6.1)

FIGURE 6.1

6.3 Beam Ports

Extending through the main concrete shield in the horizontal plane and at 45 degrees with the axis of the thermal column are two beam ports. To use this experimental facility replaceable air filled aluminum tubes will extend the permanent beam ports to the core face. When not in use the permanent ports will be plugged with high density material.

Chapter 7

7.0 Control and Instrumentation

7.1 General

The control system used in the PETR is essentially that used for the BSF. The primary changes stem from the fact that the rods enter from the bottom rather than from the top. This changes the instrumentation primarily as regards physical arrangement. Due to the bottom position for the rods an entirely new drive has been designed. The aim throughout has been a "fail-safe" system.

7.2 Instrumentation

As is the case in the BSF, the reactor instrumentation is arranged in three groups: a safety circuit, a control circuit and a linear counting circuit. The safety circuit consists of two PCP ion chambers feeding sigma amplifiers and a compensated chamber supplying a signal to the sigma bus thru a period amplifier. These three chambers control the solenoid amplifier which supplies power to the solenoid valve controlling the scram valves in the hydraulic system. In addition to its function as a period safety, the compensated ion chamber also supplies a log-N indication of a compensated chamber driving a solenoid amplifier which controls the four-way valve governing control rod position. This circuit also indicates reactor flux level. The linear counting circuit uses a fission chamber to give a counting indication on a scale and to drive a log-count rate meter.

This instrumentation, the manual control for the rods and the rod position indicators are centralized in a control desk on the balcony at the edge of the pool. This arrangement allows the operator to have all operations

under direct observation at all times thus overcoming one of the prime operational difficulties encountered at the BSF.

7.3 Rod Drives

The rod drives are entirely hydraulic. Each rod is positioned by a hydraulic cylinder and an individual hydraulic circuit. Position control is by means of a four-way three position valve. The valves are spring centered (all ports blocked) and solenoid off-set. Power is supplied by a vane type oil pump for each drive. Scram action is effected from a gas loaded accumulator acting thru pilot operated valves independent on the four-way control valve. Loss of power to the scram valve causes the bottom of the drive cylinder to be opened to the hydraulic reservoir and the top to be connected to the accumulator driving the rod down. The entire system including the cylinders are standard "off-the-shelf" components.

The dynamic response of this system remains to be investigated. In view of the large line sizes and high pressures available, however, no qualms are felt as to the adequacy of the system in this respect. It seems certain that scram times in the order of a fraction of a second can be achieved.

ELEMENTARY DIAGRAM
OF CONTROL & INSTRUMENTATION

FIG 7.2

Chapter 8

8.0 Experimental Loops

8.1 Circulating Fuel Loop

In order that experiments on a circulating fuel may be performed with this reactor, it is proposed that a loop which is entirely self-contained be inserted into the core. The loop, as presently conceived, would be a U-tube containing uranyl sulphate. This U-tube would replace two of the fuel elements in the core array. In addition to the U-tube, there would be a canned-rotor pump and a heat exchanger. The arrangement as shown in Appendix D is designed to allow a power level of 100 kilowatts in the loop.

Both the U-tube and the tubes of the heat exchanger are of double-wall construction; the inner wall, which is in contact with the uranyl sulphate, being made of stainless steel and the exterior wall, which is in contact with the cooling water, being made of aluminum. The space between these walls would be continuously monitored in order to detect leakage. The cooling water, required for the heat exchanger, is obtained from pipes which plug into the cooling water inlet openings in the grid plate. It is proposed that all of the equipment and instrumentation required for this loop would be built as a unit so that the equipment could be installed and removed while under water. When not in use the equipment would be stored in the well at the far end of the pool.

8.2 Solid Fuel Element Loops

It is conceivable that it would be desirable to run tests on solid fuel elements other than those used normally in the PETR core. In this case it would be desirable to isolate the element under test from the core proper

insofar as the cooling circuit is concerned. For this purpose two chambers, which respectively replace 4 and 9 elements of the PETR core have been proposed. The cooling water for these chambers would be supplied from pipes which plug into the PETR grid plate in a fashion similar to that used for the fluid fuel loop. The instrumentation of these chambers would be self-contained so as to make possible the underwater installation and dismantling of the loops. Further detail on this structure is shown in Appendix G.

Chapter 9

9.0 Cost Estimates

9.1 General

The estimate of costs for the PETR is based on a location at ORNL. The costs shown are in general based on current quotations from vendors or upon recent experience with comparable structures. The figures shown reflect an effort to hold costs to a minimum without jeopardizing the adequacy of the facility.

9.2 Summary of Costs

Reactor (with fuel, controls, structure, etc.)	\$57,100	
Cooling System (pumps, heat exchanger, cooling tower)	33,200	
Pool Structure	86,600	
Building	91,700	
Miscellaneous (site preparation, utilities, etc.)	25,000	
		<hr/>
		\$293,600
Contingencies	26,400	
		\$320,000
Engineering	30,000	
		<hr/>
Total		\$350,000

9.3 Discussion of Costs

It should be noted that the postulation of a site at ORNL imposes a penalty of approximately \$50,000 due to the occurrence of rock which must be excavated for foundation and pool. This is clearly shown in the details of cost set out in Appendix F.

Because of the somewhat tenuous nature of the costs of the experimental equipment which depend very strongly on the nature of their final design these costs are not included above.

It is thought, however, that \$80,000 should cover the thermal column, beam ports, instrumentation (other than reactor controls) and such things as the "loops". If it were desired to eliminate pressurized operation, the cost of the reactor should be reduced to approximately \$55,000 and the cooling system would be reduced to about \$20,000.

APPENDIX A

PRESSURE DROP AND HEAT TRANSFER DATA

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A.1.1 PRESSURE DROP IN FUEL ELEMENTS

The pressure drop in the fuel assembly (Figure 2.3) was calculated using pressure drop equations of form similar to those described in references 3, 4, and 5. The fuel assembly was divided into three separate units, the entrance region, the fuel region and the exit region. The pressure drop is presented as a series of equations (one for each region), an over-all equation and a plot of pressure drop versus velocity in the fuel section (Fig. A.1.1b).

The fuel element may be described by the following area and velocity ratios:

Flow Areas

In the 2.250" diameter entrance region	0.0276 ft ²
In the fuel section	0.0404 ft ²
In the 3.019" by 2.746" square section	0.0576 ft ²

Velocities

In the fuel section	$U = V$ ft/sec
In the 2.250" entrance section	$U = 1.464 V$ sec.
In the 3.019" by 2.746 square section	$U = 0.701 V$ "

I. Pressure Loss in the Entrance Region

The entrance region pressure drop may be thought to be composed of the contraction loss to the 2.250" pipe, plus the friction loss in the 2.2500" diameter pipe plus the expansion loss from the pipe to the square section plus the friction loss in the square section. All water properties are taken to be at an average water temperature of 100°F and friction factor data is taken from Figure 5.1 of Reference 3.

FIGURE A-1.18

A. Contraction Loss

$$\Delta P = \frac{K_c U^2 \rho}{2 g (144)}$$

$$= \frac{0.4 (1.464V)^2 62.0}{2 \times 144 \times 32.2} = 0.0057 V^2$$

B. Friction in 2.250" Pipe

$$Re = \frac{du \rho}{\mu}$$

$$= \frac{2.25 (1.464) V (67.0)}{12 \times 0.000484} = 3.50 \times 10^4 V$$

$$\Delta p = \frac{2 f U^2 \rho L}{144 \text{ Degt}}$$

$$= \frac{2 f (1.424 V)^2 62.0 \times \frac{6.76}{12}}{144 \times \frac{2.25}{12} \times 32.2} = 0.171 V^2 f$$

V	f	Δp
1	0.0060	0.00103
5	0.0045	0.0193
10	0.0036	0.0618
20	0.0030	0.2260

When this data is plotted on log-log paper it is found that the equation of a line intersecting the points has of the form

$$\Delta p = 0.00100 V^{1.79}$$

C. Expansion from the round to the square section

$$\Delta P = \frac{\rho U^2}{2g (144)} \left[1 - \left(\frac{A_1}{A_2} \right) \right]^2$$

$$= \frac{62.0 (1.464 V)^2}{2 \times 32.2 \times 144} \left[1 - \left(\frac{0.0276}{0.0576} \right) \right]^2 = 0.00390 V^2$$

D. Friction Drop in the Square Section

$$Re = \frac{\rho U D_2}{\mu}$$

$$= \frac{(62.0)(.701 V) \left[\frac{12 \times 4 \times 0.0576}{2 \times 5.765} \right]}{0.000484} = 2.16 \times 10^4 V$$

$$\Delta p = \frac{2 f U^2 \rho_L}{C D g_e (144)}$$

$$= \frac{2 f (.701 V)^2 62.0 \times 1}{12 \times 32.2 \times 144 \times \left[\frac{12 \times 4 \times 0.0576}{2 \times 5.765} \right]} = .0022 f V^2$$

V	f	Δp
1	.0068	.000015
5	.0048	.000263
10	.0041	.000905
20	.0037	.00325

These points may be described as before by the equation:

$$\Delta p = 0.000015 V^{1.79}$$

E. The total pressure drop in the entrance section of the fuel elements is described by the relation:

$$\Delta p = 0.001015 V^{1.79} + .0096 V^2$$

II. Pressure Drop in the Fuel Section

This is composed of the contraction loss plus the friction drop in the plates plus the loss due to expansion to the square section.

A. Contraction Loss

$$\begin{aligned} \Delta p &= \frac{K_c U^2 \rho}{2g (144)} \\ &= \frac{(.3) (V^2) (62.0)}{2 (32.2) (144)} = 0.00202 V^2 \end{aligned}$$

B. Friction Loss in the Fuel Plates

$$\begin{aligned} Re &= \frac{\rho U D_2}{\mu} \\ &= \frac{62.0 V \times .235}{12 \times .000484} = 2.51 \times 10^3 V \end{aligned}$$

$$\Delta p = \frac{2 f v^2 \rho L}{g D_e (144)}$$

$$= \frac{2 f (v^2) (62.0) (2.052) (12)}{(32.2) (0.235) (144)} = 2.801 f v^2$$

V	f	Δp
1	0.0100	0.0280
5	0.0075	0.526
10	0.0065	1.795
20	0.0055	6.164

This data may be expressed in the form:

$$\Delta p = 0.0290 v^{1.79}$$

C. Expansion Loss

$$\Delta p = \frac{v^2 \rho}{2 g (144)} \left[1 - \frac{A_1}{A_2} \right]^2$$

$$= \frac{62.0 v^2}{2 \times 32.2 \times 144} \left[1 - \frac{.0404}{.0576} \right]^2 = 0.00066 v^2$$

D. Total Pressure Drop in Fuel Section

$$\Delta p = 0.0290 v^{1.79} + 0.00268 v^2$$

III. Pressure Drop in the Exit Region

The pressure loss in the exit region is due to friction loss in the square section plus expansion out of the element.

A. Friction Loss

$$Re = \frac{\rho U D_e}{\mu} = 2.16 \times 10^4 V \text{ (same as in entrance region)}$$

$$\Delta p = \frac{2 f \rho U^2 L}{144 g D_e}$$

$$= \frac{2 \times \frac{2}{12} \times (.701 V)^2 \times 62.0 f}{144 \times 32.2 \times \frac{2.87}{12}} = 0.0044 f V^2$$

V	f	Δp
1	.0068	.0000299
5	.0048	.0005260
10	.0041	.0018100
20	.0037	.00650

This data may be expressed as

$$\Delta p = 0.000029 V^{1.79}$$

B. Expansion Loss

$$\Delta p = \frac{f U^2}{2 g (144)} \left[1 - \frac{A_1}{A_2} \right]^2$$

$$= \frac{62.0 (.701 V)^2}{2 g (144)} (1) = 0.00328 V^2$$

C. Total Exit Pressure Loss

$$\Delta p = 0.00328 U^2 + 0.000029 V^{1.79}$$

IV. Total Pressure Drop in the Fuel Element

The total fuel element pressure drop is shown in Figure A.1.1b and is described by the relation

$$\Delta p = 0.0303 V^{1.79} + 0.01556 V^2$$

A.1.2 PRESSURE DROP IN THE VALVE-PIPE STUB COMBINATION

The pressure drop in the 12 inch gate valve and accompanying 1.2 foot of pipe stub were calculated with equations similar to those in Reference 1. The twelve inch gate valve was taken to have a resistance equivalent to 6.8 feet of twelve inch pipe (Reference 2). The assumptions involved were:

1. Contraction from an infinite area to that of the valve.
2. Expansion from pipe flow area to infinite area of the lower plenum chamber.

A. Contraction Loss

$$\Delta p = K_c \frac{\rho U^2}{2 g (144)}$$

$$= \frac{62.0 U^2}{2 \times 32.2 \times 144} = .00268 U^2$$

B. Friction Loss

$$\Delta p = \frac{2 f U^2 \rho L}{144 g D}$$

$$= \frac{2 \times f \times 62.0 \times 8 U^2}{144 \times 32.2 \times 1} = 0.214 f U^2$$

$$R_e = \frac{\rho U D}{\mu} = 1.26 \times 10^5 U^2$$

U	f	Δp
1	.0047	.00100
5	.0033	.0176
10	.0027	.575
20	.0021	1.81

This data may be described by

$$\Delta p = 0.00100 U^{1.79}$$

C. Expansion Loss

$$\Delta p = \frac{\rho U^2}{2 g (144)} \left[1 - \frac{A_1}{A_2} \right]^2$$

$$= \frac{62 \times U^2}{2 \times 32.2 \times 144} = .0065 U^2$$

D. Total Valve Pressure Drop

$$\Delta p = 0.00100 U^{1.79} + 0.0091 U^2$$

In order to relate this pressure drop through the gate valve to the velocity in the fuel element, the limiting case of twenty-five elements operated at 1 Mw and each drawing an equal amount of water was taken. Thus:

Velocity in the valve, $U = 1.3 V$ ft/sec where V is the velocity in the fuel and:

FIGURE A-1.2a

$$\Delta p = 0.0016 V^{1.79} + 0.0154 V^2$$

This relation is plotted in Figure A.1.2a.

A.1.3 NATURAL CIRCULATION DRIVING FORCE

Since the design of the engineering test reactor necessitates the introduction of additional flow resistance in the natural circulation path, a study was made to determine maximum operating temperatures and water flow rates. The case considered consists of a reactor composed of twenty five elements arranged in a give by five array containing three control rods (which do not generate power) operating at one megawatt. The assumption is made that the central element operates at twice the average operating power and that the heat is introduced into the coolant stream in a linear fashion.

These assumptions should make the resultant flow rate a conservative estimate of the actual flow rate. The driving force is caused by the non-uniformity of density along the flow path. Thus:

$$\Delta p = \frac{(\rho_i - \rho_e) H}{2 (144)}$$

The total heat generated in the centermost element is

$$Q = \frac{2 \times 1 \text{ Mw}}{22} = \frac{3,413,000 \text{ BTU/hr}}{22} = 87.0 \text{ BTU/sec}$$

The exit temperature may be calculated through the use of the first law of thermodynamics,

$$Q = \omega C_p (T_i - T_e)$$

Returning to the previous equation the pressure difference may now be calculated if the inlet temperature is assumed.

Assuming an Inlet Temperature 100°F.

ω #/sec	T_e °F	ρ_e #/ft ³	$\rho_i - \rho_e$	Δp psi
∞	100	62.0	0	—
10	108.7	61.9	0.1	.00078
7.5	111.5	61.85	0.15	.00115
5.0	117.5	61.77	0.23	.00177
4.0	121.8	61.7	0.30	.0023
3.0	129.0	61.6	0.40	.0031
2.5	134.8	61.5	0.50	.00385
2.0	143.5	61.35	0.65	.00505
1.5	158.0	61.05	0.95	.0073
1.0	187.0	60.7	1.30	.0100

Figure A.1.3a shows a cross plot of pressure drop needed for this reactor arrangement and driving force available. The intersection determines the natural circulation flow rate for this power level.

A.1.4 SPECIFICATIONS OF PRIMARY COOLANT LOOP, SECONDARY COOLANT LOOP AND DEMINERALIZER LOOP

The primary coolant loop, secondary coolant loop and demineralizer loop are shown schematically in Figure A.1.4a. The approximate lay-out of the piping is shown in Figures A.1.5a, A.1.6a, and A.1.7a. The pressure drop and temperatures at design flow for operation as a pressurized water reactor at one megawatt are:

FIGURE A-1.3a

Fig. A-1.4a Cooling Water Diagram.

	$T_{\max}$ °F	$T_{\min}$ °F	ω #/sec	Δp	$T_{\text{air max}}$ °F
Primary Loop	170	140	31.6	9.21	—
Secondary Loop	140	110	31.6	29.9	90
Demineralizer	140	—	2.5	19.0	—

When the reactor is cooled by natural circulation the pool may be cooled by operation of the secondary coolant loop. The design conditions for operation as a natural circulation reactor are:

	$T_{\max}$ °F	$T_{\min}$ °F	ω #/sec	Δp	$T_{\text{air max}}$ °F
Secondary Loop	110	100	31.6	29.9	90
Demineralizer	110	—	2.5	19.0	—

This allows for continuous operation at 250 Kw or daytime operation (6 hours) at 1 Mw if the pool is circulated and cooled at night. The heat capacity of the pool is such that with no cooling at one megawatt operation, the pool temperature will rise at the rate of 5°F per hour.

A.1.5 PRIMARY COOLANT LOOP

The primary coolant loop is used to cool the reactor when operating as a pressurized water reactor and to supply cooling water to the solid and circulating fuel loops. The water flows from the tank below the reactor to the heat exchanger, thru the pump and pressurizer and back to the top of the reactor. Thus, in forced convection cooling, the flow is downward thru the fuel element. Provision is made for the necessary valving to by-pass a small amount of flow thru the demineralizer.

The primary coolant loop consists of the reactor, 90 feet of 4" pipe with 14 rt. angle bends, 4 valves similar to Crane Co. valve # 318, 1 valve

Fig. A-1.58 Primary Coolant Loop.

similar to Crane Co. Valve # 4437, one heat exchanger, one heat exchanger, one pressurizer and surge tank, one make-up pump, and one main circulating pump.

Design flow = 31.6 #/sec

Max. design temp. = 170 °F

Min. design ΔT = 30 °F

Pressure Drop at Design Condition

Reactor	0.1 psi
Pipe	1.5 psi
Bends	1.6 psi
Valves	1.0 psi
Heat exchanger	5.0 psi
Pressurizer	0.01 psi
Total	9.2 psi

Thus the circulating pump must deliver 31.6 #/sec at 9.2 psi. A circulating pump of design similar to Allis-Chalmers Pump Type SS-D is suggested. The make-up pump should have a capacity of 2 #/sec at 125 psi. A pump similar to Ingersoll Rand Pump #1 MRVN 5 would be suitable.

A.1.6 SECONDARY COOLANT LOOP

The secondary coolant loop is primarily used to perform one or more of three important functions:

1. Cooling of the primary loop when the reactor is operating with forced circulation cooling.
2. Cooling of the pool water during natural circulation operation.

3. Provide a water stream to diffuse the radioactive water rising from the reactor core to allow the activity to decay before reaching the surface.

The secondary loop draws water both from the top of the pool for skimming purposes and from the bottom of the pool thru the primary heat exchanger and out to a fin-fan cooler, then back to the pool thru diffusing jets mounted on the pool wall over the reactor core.

The secondary system consists of 320 ft of 4" da. pipe, 12 rt. angle bends, 7 valves similar to Crane Co. # 318, 1 valve similar to Crane Co. # 4437, one heat exchanger, one full flow filter, one fin fan cooler, and one main circulation pump.

Max. Design Flow	31.6 #/sec
Max. Design Temperature	140 °F
Max. Design ΔT	30 °F

Pressure Drop at Design Conditions

Pipe	5.0 psi
Bends	1.3 psi
Valves	1.6 psi
Filter	10.0 psi
Heat Exchanger	8.0 psi
Fin Fan Cooler	5.0 psi

Total	29.9 psi
-------	----------

Thus the circulating pump must deliver 31.6 #/sec at 29.9 psi. A circulating pump similar to Allis-Chalmers Type SS-N is suggested.

Fig. A-1.6a Secondary Coolant Loop.

A.1.7 DEMINERALIZER CIRCUIT

The demineralizer circuit consists of 80 feet of 1½" pipe, 9 rt. angle bends, 3 valves similar to Crane Co. # 440, one demineralizer bed, one filter, and a circulating pump.

Max. Design Flow 18 gpm

Max. Design Temperature 140 °F

Pressure Drop at Design Condition

Pipe 1.4 psi

Bends 0.5 psi

Valves 0.5 psi

Entrance and exit
flow to bed 7.5 psi

Filter 5.0 psi

Bed 3.6 psi

Total 18.5 psi

A circulating pump similar to Allis-Chalmers Type SS-RH with a flow of 18 gpm at 18.5 psi is suggested.

Fig. A-1.7a Demineralizer Loop.

A.2 HEAT TRANSFER

A.2.1 HEAT GENERATION RATE IN FUEL

In calculating the maximum temperature drop from fuel-element surface to coolant, the assumption that neutron flux, and hence (with a uniform fuel distribution) heat flux is distributed as a cosine curve about the center of the reactor (exact for a homogeneous bare reactor) leads to the equation:

$$Q = Q_0 \left(\cos \frac{\pi x}{\tilde{L}_x} \right) \left(\cos \frac{\pi y}{\tilde{L}_y} \right) \left(\cos \frac{\pi z}{\tilde{L}_z} \right)$$

where

Q_0 = maximum heat generation rate BTU/sec ft³

$\tilde{L}_x$ = Extrapolated length in the X direction

$\tilde{L}_y$ = Extrapolated length in the Y direction

$\tilde{L}_z$ = Extrapolated length in the Z direction

The maximum volume heat generation rate Q_0 may be determined from the equation:

$$P = Q_0 \int_{-\frac{L_x}{2}}^{\frac{L_x}{2}} \int_{-\frac{L_y}{2}}^{\frac{L_y}{2}} \int_{-\frac{L_z}{2}}^{\frac{L_z}{2}} \left(\cos \frac{\pi x}{\tilde{L}_x} \cos \frac{\pi y}{\tilde{L}_y} \cos \frac{\pi z}{\tilde{L}_z} \right) dx dy dz$$

Where P = total power produced, BTU/sec.

If the center element is assumed to produce twice as much power as the average element the extrapolated lengths are related to actual lengths by

$$\tilde{L} = 1.08 L$$

This implies that the heat generation rate at the edge of the core in the

center of each face is equal to approximately 1/10 the maximum heat generation rate at the center of the core. The ratio of maximum to average heat generation is

$$\frac{q_0}{q} = 3.16$$

Again, the limiting case is that of a reactor formed by a five by five array of fuel elements and three control rods, operating at one megawatt.

A.2.2 MAXIMUM FUEL SURFACE TEMPERATURE

Heat transfer area of each element equals 15.8 ft²/element. Thus total heat transfer area for 22 active elements equals 348 ft².

Assuming all heat generated in the reactor is generated in the fuel and must be transferred directly to the water (no axial heat conduction) the maximum surface heat flux is:

$$q_0 = \frac{3,413,000 \times 3.16}{348 \times 3600} = 8.6 \text{ BTU/sec ft}^2$$

In the central element the heat flux may be described by:

$$q = q_0 \cos \left(\frac{\pi z}{L_z} \right)$$

If the flow turbulent ($Re > 2000$, $V \geq 0.8 \text{ ft/sec}$) the heat transfer coefficient may be calculated from the familiar relation (Reference 3)

$$\frac{hD}{k} = 0.023 \left(\frac{\rho_{UD}}{\mu} \right)^{0.8} \left(\frac{\mu C_p}{k} \right)^{0.40}$$

Evaluating the physical quantities at a mean coolant temperature of 150°F

$$h = 436 V^{0.8}$$

where:

$$h = \text{BTU/hr ft}^2 \text{ } ^\circ\text{F}$$

$$V = \text{ft/sec}$$

and since

$$q_o = h (T_f - T_c)$$

The previous equations may be combined in the form:

$$q_o = 8.6 \cos \left(\frac{\pi z}{\tilde{L}_z} \right) = \frac{436 V^{0.8}}{3600} (T_f - T_c)$$

or

$$T_f - T_c = \frac{70.5}{V^{0.8}} \cos \left(\frac{\pi z}{\tilde{L}_z} \right)$$

Figure A.2.2a is a plot of maximum temperature drop versus velocity.

At extremely low velocities the heat transfer coefficient may be calculated from the Colburn equation (Reference 3) for natural circulation heating of water in vertical tubes.

$$h = 0.42 T (\Delta T)^{1/3}$$

where

$$h = \text{BTU/hr ft}^2 \text{ } ^\circ\text{F}$$

$$T = \text{average film temperature } ^\circ\text{F}$$

$$\Delta T = \text{average temperature difference between coolant and fuel plate, } ^\circ\text{F.}$$

Evaluating the physical properties of water at a mean coolant temperature of 200°F with an average temperature difference of 80°F , one has

$$h = 360 \text{ BTU/hr ft}^2 \text{ } ^\circ\text{F}$$

Thus,

$$q_o = 8.6 \cos \frac{\pi z}{\tilde{L}_z} = \frac{360}{3600} (T_f - T_c)$$

or

$$(T_f - T_c) = \frac{8.6 \times 3600}{360} \cos \left(\frac{\pi z}{\tilde{L}_z} \right) = 86 \cos \left(\frac{\pi z}{\tilde{L}_z} \right)$$

If use is made of the first law of thermodynamics, the coolant temperature as a function of length may be determined by:

$$C_p \omega (T_z - T_1) = \int_{\frac{-L_z}{2.16}}^z A q_o \cos \left(\frac{\pi z}{\tilde{L}_z} \right) dz$$

where $A = \text{ft}^2$ heat transfer area per ft of length.

The results of these equations for both laminar and turbulent flow are plotted in Figure A.2.2b for the centermost element, together with experimental measurements of maximum fuel plate temperature on the centermost element of the Oak Ridge Bulk Shielding Reactor operating at one megawatt with a 28 element matrix (Reference 6).

Figure A.2.2b predicts that, with natural circulation flows as low as one pound per second (0.4 ft/sec) in the centermost element, the maximum temperature rise is only 136°F above the coolant inlet temperature. Since the boiling point of water is 238°F at 24 psia (under a 22 ft. head of water) the maximum allowable coolant inlet temperature is 102°F . However, this is not really as bad as it seems. If the reactor is compared to its mechanical counterpart a natural circulation boiler with vertical tubes-it is found (Reference 3) that a temperature difference as small as ten degrees F above the boiling point increases the coefficient of heat transfer by a factor of five.

A cross plot of maximum temperature difference between the surface of the fuel plate to the inlet water temperature versus coolant flow rate is shown in Figure A.2.2c for the assumed maximum case of 17 active elements producing one megawatt with a maximum to average heat flux ratio of 3.16.

FIGURE A-2.2a

TEMPERATURE DIFFERENCE °F

FIGURE A-2.2b

FIGURE A-2.2c

A.3 NOMENCLATURE FOR APPENDIX

A	= Area ft ²
C _p	= Specific heat BTU/#°F
D	= Diameter
H	= Head ft.
K _c	= Contraction coefficient
L	= Length-feet
Q	= Heat generation BTU/sec
Re	= Reynolds Number
T	= Temperature - degrees F
V	= Velocity in fuel sections of reactor ft/sec
U	= Velocity - ft/sec
X	= Length in "x" direction ft
Y	= Length in "y" direction ft
Z	= Length in "z" direction ft
f	= Friction factor
g	= Acceleration of gravity - 32.2 ft/sec
h	= Coefficient of heat transfer - BTU/hr ft ² °F
k	= Thermal conductivity; BTU/ft °F
c	= Pressure psi
q	= Heat flux BTU/sec ft ²
Δ	= An increment
μ	= Viscosity #/ft sec
ρ	= Density #/ft ³
w	= Mass flow #/sec

SUBSCRIPTS

c	= coolant
e	= exit
E	= equilivent
f	= fuel
i	= inlet
o	= max. value
p	= constant pressure
x	= in "x" direction
y	= in "y" direction
z	= in "z" direction

A.4 REFERENCES

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APPENDIX B

GAMMA AND NEUTRON HEATING OF STRUCTURAL MATERIALS

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B.1.1 HEAT GENERATION IN THE PRESSURE VESSEL AND GRID PLATE

Heat is generated by the absorption and scattering of gamma rays and neutrons in structural members surrounding the reactor. The rate of heat generation was determined for the principal core structural material aluminum and compared with the available experimental data. For aluminum, the neutron cross sections are so low that most of the heating is caused by the absorption and scattering of gamma rays.

The gamma spectrum at the core face is shown in Figure B.1.1a for one megawatt operation. (Ref. 1) The heat generation may be calculated from:

$$G_{\gamma} = \int_0^{\infty} \mu_e \Gamma(E) E dE \times 1.545 \times 10^{-8}$$

$$\approx \sum_n \mu_e \Gamma_n(E) E_n \Delta E_n \times 1.545 \times 10^{-8}$$

at the surface of the aluminum next to the core face.

E(Mev)	$\Gamma(E)$ (photons/cm ² Mev sec)	$\mu_e \frac{\text{cm}^2}{\text{gram}}$	G_{γ} BTU/hr ft ³
$\frac{1}{4}$	9.9×10^{13}	0.03	9,450
$\frac{1}{2}$	4.95×10^{13}	0.29	9,080
1	1.98×10^{13}	0.028	14,100
2	1.15×10^{13}	0.0225	13,100
3	3.47×10^{12}	0.021	5,490
4	1.32×10^{12}	0.02	2,660
5	5.28×10^{11}	0.0195	1,300
6	2.31×10^{11}	0.019	660
7	1.65×10^{11}	0.0185	540
Total			56,380

FIGURE B-1.1a

The neutron spectrum at the core face for one megawatt operation is shown approximately in Figure B.1.1b (Ref. 1). The heat generated by neutron reactions is thought to be negligible, however a sample calculation of energy deposited during the slowing down process is shown:

$$G = \int_0^{\infty} \delta E_n \phi_n \sum_s (E_n) dE_n \times 1.545 \times 10^{-8}$$

$$\approx \sum \delta E_n \phi(E_n) \sum_s \Delta E_n \times 1.545 \times 10^{-8}$$

$$\delta = \frac{\text{ave. energy lost per collision}}{\text{initial energy}}$$

Assuming isotropic, elastic scattering in the center of mass coordinate system the average energy loss may be calculated from

$$\delta = 1 - \left(\frac{M-m}{M+m} \right)^2$$

where

M = mass of target nucleus

m = mass of neutron

E (Mev)	$\phi(E)$ $\frac{\text{neutrons}}{\text{cm}^2\text{-sec Mev}}$	$\sum_s$ 1/cm	δ	G BTU/hr ft ³
1	1.81×10^{12}	0.067	0.069	132
2	1.20×10^{12}	0.067	0.069	171
3	8.5×10^{11}	0.067	0.069	184
4	4.5×10^{11}	0.067	0.069	129
5	2.48×10^{11}	0.067	0.069	89
6	1.49×10^{11}	0.067	0.069	64
7	0.9×10^{11}	0.067	0.069	45

FIGURE B-1.1b

Thus

$$G_n = 814 \text{ BTU/hr ft}^2$$

In Figure B.1.1c, the heat generation rate is plotted vs. inches of water between core face the aluminum specimen.

B.2 MAXIMUM SURFACE TEMPERATURE IN ALUMINUM STRUCTURAL PLATE

It is of interest to consider the magnitude and location of the maximum temperature above coolant temperature of an aluminum plate in a radiation field. If the heating is due primarily to gamma capture, as demonstrated in Appendix B.1.1, the heat generation may be described by the equation:

$$G = G_0 e^{-\mu x}$$

where G_0 is the local heating on the face nearest the reactor and μ is the energy absorption cross section.

For a plate, the equation for heat conduction with internal heat generation is of the form:

$$G(x) + k \nabla^2 T = 0$$

The solution of this equation with boundary conditions similar to those shown in Figure B.2.1a is

Figure B.2.1a.

$$T = T_1 + \frac{G_0}{K \mu^2} + \left[\frac{T_2 - T_1 + \frac{G_0}{K \mu^2} (e^{-\mu l} - 1)}{l} \right] x - \frac{G_0}{\mu^2} K e^{-\mu x}$$

The heat flux at either side is proportional to the temperature gradient at that surface.

$$q = -k \left| \frac{dT}{dx} \right| = h (T_{Al} - T_c) = \int_0^l G(x) dx$$

FIGURE B-1.1c

Thus

$$q(x=0) = k \left[T_2 - T_1 + \frac{G_o}{K\mu^2} (e^{-\mu \ell} - 1) + \frac{G_o}{\mu K} \right]$$

$$q(x=\ell) = k \left[-T_2 + T_1 - \frac{G_o}{K\mu^2} (e^{-\mu \ell} - 1) - \frac{G_o}{\mu K} e^{-\mu \ell} \right]$$

For free convection cooling, the following heat transfer coefficient is often used (Reference 3) for vertical plates more than one ft. high;

$$h = 0.128 \left(\frac{k_f^2 \rho_f^2 c_p \beta_f g (T_{Al} - T_c)}{\mu_f} \right)^{1/3}$$

The result of combining equations is plotted in Figures B.2.1b and B.2.1c.

FIGURE B-2.1b

FIGURE B-2.1c

SYMBOLS

C_p	= specific heat BTU/# °F
E	= Energy Mev
G	= Heat generation rate BTU/hr ft ³
T	= Temperature °F
g	= Acc. of gravity ft/sec ²
h	= Film coefficient BTU/hr ft ² °F
k	= Conductivity BTU/hr ft °F
l	= Plate thickness FC
q	= Heat flux BTU/hr ft ²
μ_e	= Gamma macroscopic absorption coefficient 1/cm
x	= Distance in "x" direction FC
β	= Volume thermal coefficient of expansion 1/°F
Δ	= Increment
δ	= Ratio of energy deposited to total energy
Σ	= Summation of terms
Σ	= Macroscopic cross section, 1/cm
Γ	= Gamma flux photons/sec cm ²
ρ	= Density #/ft ³
μ	= Viscosity # hr/ft
ϕ	= neutron flux <u>neutrons</u> sec cm ²

SUBSCRIPTS

c	= Coolant
f	= Film

n = Neutron
o = maximum value
 γ = Gamma

REFERENCES

1. Research Reactors, TID-5275.
2. Gamma Heating Measurements in the BSF, F. T. Binford et al, ORNL CF-56-3-72.
3. Heat Transmission, McAdams, McGraw-Hill Book Co., Inc. 1942.

APPENDIX C

NUCLEAR CALCULATIONS

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C. Physics - Criticality

The critical size of the reactor has been determined for five basic core configurations. Configuration A consists of a solid array of MTR type fuel elements. Configurations B and C contain solid fuel element test loops plus MTR type fuel elements. Configuration D consists of a liquid fuel test loop plus MTR type fuel elements. Configuration E consists of a graphite center section surrounded by MTR type fuel elements.

All configurations of the reactor are water-moderated and cooled. The test loops are located centrally in the core and provide a high neutron flux region for the study of both liquid fuels and solid fuel elements under various conditions.

The critical conditions for specific core configurations have been investigated to provide estimates of fuel requirements. Space has been provided for adding additional fuel elements. This additional fuel plus a choice of several control rod positions allows a wide variety of core configurations to be assembled.

The ORACLE has been used for all critical computations. A three group, three region code ⁽¹⁾ was employed with modifications to suit the various core configurations. Only two neutrons energy groups were used in the computations because of the high enrichment in U-235. The scarcity of U-238 in the core results in negligible absorption in the intermediate energy ranges and hence a separate flux determination in the intermediate range was not necessary.

For purposes of computation, the reactor cores were considered to be cylinders. Knowing the critical radius of a cylindrical core, the critical mass was computed. This mass was considered to be the critical mass for the

actual core configuration. It is realized that some error is introduced because the shape of the fuel elements prohibit the formation of a true cylindrical core. This error, however, is considered small for the configurations studied.

C.1 CORE CONFIGURATION A

Configuration A consists of an assembly of solid fuel elements with an infinite water reflector. The general shape of the core is shown in Figure C.1a. A description of the fuel elements, hereafter called standard fuel elements, is found in Figure 2.3.

A unit cell in the core was taken to be a volume enclosing the fuel bearing section of one fuel element plus a thin sheath of water. The dimensions of the unit cell are $24 \times 3.000 \times 3.075$ inches.

The volume of the cell is thus 3617 cm^3 . The average density of the various materials in a cell is as follows:

Aluminum	- 0.991 g/cm^3
H_2O	- 0.633 g/cm^3
U-235	- 0.0387 g/cm^3

The aluminum to water volume ratio is 0.580.

C.1.1 Region I

Region I is the fuel bearing section of the reactor. For an aluminum to water volume ratio of 0.580, the following nuclear constants are applicable⁽²⁾:

$$\begin{aligned} D \text{ (thermal)} &= 0.247 \text{ cm} \\ D \text{ (fast)} &= 1.29 \text{ cm} \\ \tau &= 57 \text{ cm}^2 \end{aligned}$$

Core Configurations
Fig. C-1a

FIGURE C-1.1a

Absorption in the fast neutron group was assumed to be negligible and hence equal to zero. The removal cross section from the fast to thermal groups is thus equal to ⁽³⁾:

$$\sum_x = \frac{D(\text{fast})}{\tau} = \frac{1.29}{57} = 0.0226 \text{ cm}^{-1}$$

The total thermal absorption cross section was computed as follows:

$$\sum_a^{\text{Total}}(\text{thermal}) = \frac{V_{\text{Al}}}{V_{\text{Total}}} \sum_a^{\text{Al}} + \frac{V_{\text{H}_2\text{O}}}{V_{\text{Total}}} \sum_a^{\text{H}_2\text{O}} + N^{\text{U-235}} \sigma_a^{\text{U-235}}$$

Using ⁽²⁾ $\sum_a^{\text{Al}} = 0.0115 \text{ cm}^{-1}$

$$\sum_a^{\text{H}_2\text{O}} = 0.0197 \text{ cm}^{-1}$$

$$\sum_a^{\text{Total}}(\text{thermal}) = 0.0835 \text{ cm}^{-1}$$

Correcting for a Maxwell-Boltzmann average, the result is:

$$\sum_a^{\text{Total}}(\text{thermal}) = 0.0740 \text{ cm}^{-1}$$

Similarly:

$$V \sum_f = \eta \sum_a^{\text{U-235}} = 0.123 \text{ cm}^{-1}$$

C.1.2 Region II

Region II is the reflector section and consists of pure light water.

For light water, the following nuclear constants were used ⁽⁴⁾:

$$D(\text{thermal}) = 0.160 \text{ cm}$$

$$D(\text{fast}) = 1.19 \text{ cm}$$

$$\tau = 31.4$$

$$\sum_x = 0.0379 \text{ cm}^{-1}$$

$$\sum_a(\text{thermal}) = 0.0197 \text{ cm}^{-1}$$

$$V \sum_f = 0.0$$

C.1.3 Critical Size

The geometric buckling was considered to be identical for both the fast and thermal groups of neutrons. Since the ORACIE performs a one dimensional calculation, it was necessary to calculate an effective one dimensional buckling. For this reactor, the radius was varied while the height was held constant. Thus, the bare buckling in the axial direction was desired. Using the following formula:

$$B^2 = \left(\frac{\pi}{H^1} \right)^2$$

an effective height (H^1) must be determined in order to determine the buckling. H^1 can be calculated as follows ⁽⁵⁾:

$$H^1 = H \text{ (actual)} + 2 \delta$$

where

$$\delta = \left(\frac{D \text{ (core)}}{D \text{ (reflector)}} \right)^{\text{fast}} M^{\frac{1}{2}} \text{ (reflector)}$$

$$M \text{ (reflector)} = \text{migration area} = L^2 + \tau$$

Thus

$$\delta = \frac{1.29}{1.19} (8.12 + 31.4)^{\frac{1}{2}} = 6.79 \text{ cm}$$

And

$$H^1 = 61.0 + 13.6 = 74.6 \text{ cm}$$

Hence

$$B^2 = \left(\frac{3.14}{74.6} \right)^2 = 0.00177 \text{ cm}^{-2}$$

Using this value for the axial buckling and the nuclear constants listed for regions I and II, the multiplication constant, k , was computed for various masses of U-235 (see Figure C.1.3a. For $k = 1$, the critical mass was 2270 grams.

FIGURE C-1.3a

FIGURE C-1.4a

CORE CONFIGURATION A

AVERAGE FLUXES

	THERMAL	FAST
I	0.233	0.694
II	0.281	0.199
III	0.093	0.043

NET CURRENTS (Positive Outward)

17 cm	-0.143	1.00
24 cm	0.052	0.23

C.1.4 Flux Plots

Plots of both the fast and thermal fluxes are shown in Figure C.1.4a. The fluxes have been normalized such that full scale on this graph is equivalent to $3.51 \times 10^7 \frac{\text{neutrons}}{\text{sec-cm}^2\text{-watt}}$

C.2 CORE CONFIGURATION B

Configuration B consists of a small solid element test loop surrounded by sufficient fuel elements used in configuration A to sustain criticality. The test loop contains one test fuel element, cooling water, and a cylindrical stainless steel containing vessel. The portion of the loop in the core occupies a volume vacated by four central fuel elements in Configuration A. A description of the small test loop is found in Appendix G.

For the purpose of a sample calculation, the test element in the loop was considered to be identical with the standard fuel elements. The general shape of the core is shown in Figure C.1a.

C.2.1 Region I

Region I consists of the small test loop and the surrounding water occupying the four center cells of the reactor. The test element in the loop constitutes 25% of this volume. The remaining 75% of region I contains stainless steel and water in a volume ratio of steel to water of 0.192.

For convenience in the criticality calculations, the nuclear constants for region I were obtained by considering the volume as homogeneous matter. Although this procedure alters the shape of the flux plots, the effect on critical mass should be small.

The nuclear constants of region I were obtained by computing the average values of the test element nuclear constants and the steel-water nuclear con-

FIGURE C-2.1a

stants. The constants for the test element are identical with those constants listed for region I, core configuration A. The nuclear constants for steel-water mixtures are plotted in Figure C.2.1a. A volume-averaging procedure was used. For example:

$$\begin{aligned} \text{Constant (average)} &= 0.250 \times \text{constant (element)} \\ &+ 0.750 \times \text{Constant (steel-water mixture)}. \end{aligned}$$

Using this averaging procedure, the following nuclear constants were obtained:

$$\begin{aligned} D \text{ (thermal)} &= 0.193 \text{ cm} \\ D \text{ (fast)} &= 1.25 \text{ cm} \\ \tau &= 40.9 \text{ cm}^2 \\ \Sigma_x &= 0.0306 \text{ cm}^{-1} \\ \Sigma_a \text{ (thermal)} &= 0.0524 \text{ cm}^{-1} \\ V \Sigma_f \text{ (thermal)} &= 0.0309 \text{ cm}^{-1} \end{aligned}$$

The equivalent radius of region I was assumed to be 8.71 cm. A cylinder with this radius will have the same volume as the four cells displaced by the test loop.

C.2.2 Region II

Region II consists of the standard elements surrounding the test loop. The nuclear constants used for this region are identical with those listed in Section C.1.1.

C.2.3. Region III

Region III is the reflector for the reactor and consists of light water. Nuclear constants for light water are listed in Section C.1.2.

C.2.4 Critical Size

Using the value of 0.00177 cm^2 for the axial buckling and the nuclear constants listed for regions I, II, and III, the multiplication constant, k , was computed for various masses of U-235 (see Figure C.2.4a). The critical mass was 3750 grams.

This mass of U-235 is the sum of the uranium contained in the test loop plus that contained in the surrounding elements.

C.2.5 Flux Plots

Plots of both the fast and thermal neutron fluxes are shown in Figure C.2.5a. The fluxes have been normalized such that full scale on the graph is equivalent to $1.77 \times 10^7 \frac{\text{neutrons}}{\text{sec-cm}^2\text{-watt}}$.

C.3 CORE CONFIGURATION C

Configuration C consists of a large solid element test loop surrounded by sufficient fuel elements used in configuration A to sustain criticality. The test loop contains four test fuel elements, cooling water, and a cylindrical stainless steel containing vessel. The portion of the loop in the core occupies a volume vacated by nine central fuel elements in configuration A.

For the purpose of a sample calculation, the test elements in the loop were considered to be identical with the standard fuel elements. The general shape of the core is shown in Figure C.1a. A description of the large test loop is found in Appendix G.

C.3.1 Region I

Region I consists of the large test loop and the surrounding water occupying the nine center cells of the reactor. The four test elements in the loop constitute 44.4% of this volume. The remaining 55.6% of region I contains

FIGURE C-2.4a

FIGURE C-2.5a

CORE CONFIGURATION B

AVERAGE FLUXES

	THERMAL	FAST
I	0.434	0.839
II	0.278	0.819
III	0.223	0.143

NET CURRENT (Positive Outward)

RADIUS	THERMAL	FAST
8.7 cm	0.122	-0.619
23.7 cm	-0.184	1.19
37.7 cm	-0.053	.094

stainless steel and water in a volume ratio of steel to water of 0.160. This ratio was obtained assuming the thickness of the stainless steel vessel to be $\frac{1}{4}$ inch.

Using the same averaging procedure used in section C.2.1, the following nuclear constants were obtained and are pertinent for this region:

$$D \text{ (thermal)} = 0.206 \text{ cm}$$

$$D \text{ (fast)} = 1.26 \text{ cm}$$

$$\tau = 44.6 \text{ cm}^2$$

$$\sum_x = 0.0283 \text{ cm}^{-1}$$

$$\sum_a \text{ (thermal)} = 0.0557 \text{ cm}^{-1}$$

$$1) \sum_f \text{ (thermal)} = 0.0546 \text{ cm}^{-1}$$

The equivalent of region I was assumed to be 13.0 cm. A cylinder with this radius will have the same volume as the nine cells displaced by the test loop.

C.3.2 Region II

Region II consists of the standard fuel elements surrounding the test loop. The nuclear constants used for this region are identical with those in section C.1.1.

C.3.3 Region III

Region III is the reflector for the reactor and consists of light water. Nuclear constants for light water are listed in section C.1.2.

C.3.4 Critical Size

Using the value of 0.00177 cm^2 for the axial buckling and the nuclear constants listed for regions I, II and III, the multiplication constant, k , was computed for various masses of U-235 (See Figure C.3.4a). The critical

FIGURE C-3.4a

mass is 4150 grams. This mass of U-235 is the sum of the uranium contained in the large test loop plus that contained in the surrounding elements.

C.3.5 Flux Plots

Plots of both the fast and thermal neutron fluxes are shown on Figure C.3.5a. The fluxes have been normalized such that full scale on the graph is equivalent to $1.48 \times 10^7 \frac{\text{neutrons}}{\text{sec-cm}^2\text{-watt}}$.

C.4 CORE CONFIGURATION D

Configuration D consists of a liquid fuel test loop surrounded by standard fuel elements used in configuration A. The test loop contains an aqueous fuel solution in a double walled stainless-aluminum vessel. A description of the loop plus its various components is given in Appendix G. The core portion of the test loop occupies a volume vacated by two central fuel elements in configuration A. The general shape of the core is shown in Figure C.1a.

C.4.1 Region I

Region I consists of the liquid test loop plus the surrounding water occupying the two center cells of the reactor. The materials constituting the test loop are listed in Appendix G together with their average densities. For convenience in the critical calculations, region I was considered to be a homogeneous mixture. $D(\text{thermal})$, $D(\text{fast})$, γ , and $\sum_x$ were assumed to be dependent only on the aluminum to water volume ratio.

The following nuclear constants are applicable for region I:

$$D(\text{thermal}) = 0.194 \text{ cm}$$

$$D(\text{fast}) = 1.23 \text{ cm}$$

FIGURE C- 3.5a

CORE CONFIGURATION C

AVERAGE FLUXES

	THERMAL	FAST
I	0.442	0.936
II	0.286	0.823
III	0.236	0.151

NET CURRENT (Positive Outward)

RADIUS	THERMAL	FAST
13 cm	0.114	-0.272
25 cm	-0.194	1.25
39 cm	0.056	0.099

$$\begin{aligned}\tau &= 40.8 \text{ cm}^2 \\ \Sigma_x &= 0.0301 \text{ cm}^{-1} \\ \Sigma_a(\text{thermal}) &= 0.0938 \text{ cm}^{-1} \\ \sqrt{\Sigma_f(\text{thermal})} &= 0.149 \text{ cm}^{-1}\end{aligned}$$

For calculation purposes, the equivalent radius of the liquid loop was assumed to be 6.1 cm.

C.4.2 Region II

Region II consists of the standard elements surrounding the liquid loop. The nuclear constants used for this region are identical with those listed in Section C.1.1.

C.4.3 Region III

Region III is the reflector for the reactor and consists of light water. Nuclear constants for light water are listed in section C.1.2.

C.4.4 Critical Size

Using the value of 0.00177 cm^2 for the axial buckling and the constants listed for regions, I, II, and III, the multiplication constant k was computed for various masses of U-235 (see Figure C.4.4a). The critical mass = 2150 grams. This mass of U-235 is the sum of the uranium contained in the liquid loop (330 grams) plus that contained in the surrounding elements.

C.4.5 Flux Plots

Plots of both the fast and thermal neutron fluxes are shown on Fig. C.4.5a. The fluxes have been normalized such that full scale on the graph is equivalent to $3.48 \times 10^7 \frac{\text{neutrons}}{\text{sec-cm}^2\text{-watt}}$.

FIGURE C-4.4a

FIGURE C-4.5a

CORE CONFIGURATION D

AVERAGE FLUXES

	THERMAL	FAST
I	0.315	0.952
II	0.218	0.634
III	0.165	0.105

NET CURRENT (Positive Outward)

RADIUS	THERMAL	FAST
6.1 cm	0.048	0.347
17.6 cm	-0.136	0.945
31.6 cm	0.039	0.068

C.5 CORE CONFIGURATION E

Configuration E consists of a homogeneous core of graphite surrounded by standard fuel elements. The graphite occupies a volume vacated by four central fuel elements in configuration A. The general shape of the core is shown in Figure C.1a.

C.5.1 Region I

Region I consists of the graphite core of the reactor. Assuming a graphite density of 1.67 g/cm^3 , the following nuclear constants are pertinent for region I:

$$D(\text{thermal}) = 0.830 \text{ cm}$$

$$D(\text{fast}) = 1.11 \text{ cm}$$

$$\tau = 350 \text{ cm}^2$$

$$\sum_x = 0.00318 \text{ cm}^{-1}$$

$$\sum_a (\text{thermal}) = 0.000238 \text{ cm}^{-1}$$

$$\sum_f (\text{thermal}) = 0.0$$

The radius equivalent of region I was assumed to be 8.71 cm.

C.5.2 Region II

Region II consists of the standard elements surrounding the graphite. The nuclear constants used for this region are identical with those listed in Section C.1.1.

C.5.3 Region III

Region III is the reflector for the reactor and consists of light water. Nuclear constants for light water are listed in Section C.1.2.

C.5.4 Critical Size

Using the value of 0.00177 cm^2 for the axial buckling and the nuclear

FIGURE C-5.4a

constants listed for regions I, II, and III, the multiplication constant, k , was computed for various masses of U-235 (see Figure C.5.4a). The critical mass is 2960 grams.

C.5.5 Flux Plots

Plots of both the fast and thermal neutron fluxes are shown in Fig. C.5.5a. The fluxes have been normalized such that full scale on the graph is equivalent to $2.40 \times 10^7 \frac{\text{neutrons}}{\text{sec-cm}^2 \text{-watt}}$.

C. Physics - Control

The physics study of the PETR with respect to reactor control has been restricted to control rod effectiveness, reactivity worth of the vacant rod channels, and the effect of the liquid test loop on the over-all control of the reactor. The temperature coefficient of the solid fuel elements is assumed to be similar to that measured at the BSF (approximately $-8 \times 10^{-5} \Delta k/^\circ\text{C}$).

Core Configuration E
Fig. C-5.5a

C.6 ROD EFFECTIVENESS

The regulating and safety rods are identical in physical shape but differ in material content. It is desirable to restrict the worth of the regulating rod so that prompt criticality is impossible to achieve by withdrawal of the regulating rod only. Thus a relatively low absorbent material, stainless steel, was chosen for construction of this rod. The remaining rods (safety rods) are required to control the large excess reactivity in the core and hence were constructed of highly absorbent material.

The reactivity worth of the control rods were computed with the aid of the ORACLE for core configuration A (section C.1). The worth of a rod was computed for a central location in the core and then the worth at other non-central positions was determined by normalizing to the square of the neutron fluxes at each position as follows:

$$\text{Worth (r = R)} = \text{Worth (r = o)} \left[\frac{\sum_a (\text{thermal})}{\sum_a (\text{total})} \frac{\phi^2 (r = R) (\text{thermal})}{\phi^2 (r = o) (\text{thermal})} + \frac{\sum_a (\text{fast})}{\sum_a (\text{total})} \frac{\phi^2 (r = R) (\text{fast})}{\phi^2 (r = o) (\text{fast})} \right]$$

C.6.1 Regulating Rod

The regulating rod consists of a stainless steel sleeve with dimensions as shown in Figure C.6.1a. The sleeve is 0.087 inches thick and occupies a volume of 204 cm³. The volume of the water void caused by the rod channel is 1160 cm³. Hence the volume ratio of steel to water is:

$$\frac{\text{steel}}{\text{water}} = \frac{204}{1160-204} = 0.213$$

FIGURE C-6.1A

The nuclear constants appropriate for this steel to water ratio are:

$$\begin{aligned}
 D \text{ (thermal) } &= 0.177 \text{ cm} \\
 D \text{ (fast) } &= 1.23 \text{ cm} \\
 \tau &= 35.8 \text{ cm}^2 \\
 \Sigma_x &= 0.0344 \text{ cm}^{-1} \\
 \Sigma_a \text{ (thermal) } &= 0.0476 \text{ cm}^{-1} \\
 \sqrt{\Sigma_f} \text{ (thermal) } &= 0.0
 \end{aligned}$$

For calculation purposes, the water void and contained rod are assumed to be contained in a circular cylindrical volume whose volume is equivalent to the actual physical volume. The radius of this equivalent cylinder is 2.46 cm.

The reactivity worth of the regulating rod-vs-distance from the center of the core was plotted in Figure C.6.1b for core configuration A.

C.6.2 Safety Rods

Each safety rod consists of an aluminum shell filled with boron carbide powder. The outside dimensions of the safety rods are identical with the regulation rod. Each safety rod contains 695 cm^3 of B_4C powder (347 grams of boron and 1420 grams of carbon).

The thermal macroscopic absorption cross section for the rod is assumed to be due to boron alone and is equal to 18.6 cm^{-1} . The rod is thus considered to be a "black" absorber with an actual radius of 2.42 cm. The effective radius of the rod is (b) $r_{\text{eff}} = r_o e^{-(0.82 \times 0.741/2.42)} = 1.88 \text{ cm}$

The reactivity worth of a safety rod-vs-distance from the center of the core is plotted in Figure C.6.2a for core configuration A.

FIGURE C-6.1b

FIGURE C-6.2

C.6.3 Water Void

Both the regulating and safety rods travel in special fuel elements having longitudinal holes 1-1/8 by 2-5/8 inches. When a rod is withdrawn from the core, the resulting void fills with water. The effect of this water void on reactivity was calculated and plotted in Figure C.6.3a for core configuration A.

C.7 LIQUID LOOP TEMPERATURE COEFFICIENT

The temperature coefficient of the liquid fuel loop was determined by computing the multiplication constant k of the reactor at various temperatures.

For the temperature range of 68°F - 250°F, it was determined that only $\sum_a$ and $\nu \sum_f$ vary with temperature through density and thermal cross section changes. Changes in D (thermal), D (fast), γ , and $\sum_x$ were negligible. Size changes of the solid components were also negligible.

Figure C.7.1a shows the change in reactivity of core configuration D vs temperature of the loop. The average temperature coefficient is $-7 \times 10^{-5} \frac{k}{^\circ F}$.

C.8 EXCESS REACTIVITY REQUIREMENTS

The cold-clean uranium requirements for each of the five basic core configurations are listed in Figure C.8.1a. Cold-clean is defined as that loading of the reactor which will maintain criticality with all rods withdrawn from the core, with the reactor at room temperature (68°F), and with no neutron poisons present. The estimated excess reactivity required to allow operation at elevated temperatures, with poisons present, and under various experimental conditions is shown in Figure C.8.1b. Estimated operational uranium requirements

FIGURE C-6.8a

FIGURE C-7. /a

were calculated with the aid of k vs V-235 mass plots for each basic core configuration and are included in Figure C.8.1a.

Figure C.8.1a

URANIUM 235 REQUIREMENTS (grams)

Core Configuration	Calculated Critical Mass	Cold-Clean* Critical Mass	Operational** Loading
A	2270	2480	2860
B	3750	4050	4600
C	4150	4500	5200
D	2150	2380	2800
E	2940	3180	3630

* Includes reactivity loss of four rod channels.

** Includes excess reactivity listed in Figure C.8.1b.

EXCESS REACTIVITY REQUIREMENTS

Core Configuration	Fission Product Poisoning	Fuel Burn-up	Temperature Coefficient	Experimental Losses	Total Requirements
A-Pressurized	2.0%	0.5%	1.2%	0	3.7%
A-Non-Pressurized	2.0%	0.5%	0.3%	0.7%	3.5%
B	2.0%	0.5%	0.3%	0.7%	3.5%
C	2.0%	0.5%	0.3%	0.7%	3.5%
D	2.0%	0.5%	0.8%	0.7%	4.0%
E	2.0%	0.5%	0.3%	0.7%	3.5%

The above estimates of excess reactivity requirements are considered sufficient for operation of the reactor for a one year period. Average operation of the reactor was considered to be four hours per day at one megawatt.

Figure C.8.1b

C.9 SYMBOLS

B^2	Geometric buckling
D (thermal)	Average thermal neutron diffusion coefficient
D^S	
D (fast)	Average neutron diffusion coefficient from fission to thermal energies
D^F	
H	Height of reactor
k	Multiplication constant
L	Diffusion length
M	Migration area
N	Atomic density
V	Volume
δ	Reflector savings
λ_{Tr}	Transport mean free path
ν	Average number of neutrons released per neutron fission
η	Average number of neutrons released per neutron capture in the fuel
τ	Fermi Age from fission to thermal energy
σ_a	Microscopic absorption cross section
Σ_a	Macroscopic absorption cross section
Σ_f	Macroscopic fission cross section
Σ_x	Macroscopic removal cross section from fast to thermal energy groups

C.10 REFERENCES

1. R. R. Bate et al, "Description and Operation Manual for the Three Group, Three Region Reactor Code for ORACLE", ORNL CF-55-1-76, (55).
2. H. L. McMurry, "Nuclear Constants for the MTR as a Function of Fuel Content, and Al/H₂O Ratio", IDO-16127, (53).
3. D. K. Holmes and R. V. Meghreblian, "Notes on Reactor Analysis", Part II, ORNL CF-54-8-88, (54).
4. J. W. Webster, "Reactivity Effect of Reducing the Al/H₂O Ratio in the MTR Core", IDO - 16133, (53).
5. Personal communication from E. Silver, ORNL.
6. "The Reactor Handbook", Volume I, AECD - 3645, (55), Table 1.6.4.

APPENDIX D

BIOLOGICAL SHIELDING

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D.1 BASIS FOR SHIELD DESIGN

The shielding of the proposed reactor has been designed to attenuate by absorbing the high intensity neutron and gamma radiations arising in the core and reflector regions at one megawatt heat generation. In general, the shielding materials are so arranged that the dose rate is one-tenth of biological tolerance in those areas which are accessible to personnel during full power operation. The biological tolerance dose rate has been taken to be 7.5 mrem per hour. Somewhat higher than 1/10 tolerance dose rate will necessarily be permitted in some regions around the experimental facilities when in use, but it is intended that these regions be sufficiently well defined to constitute no biological hazard. The radiation level outside the main shield will be low enough to allow operation of sensitive research instruments with little additional shielding.

Experimental data from the Bulk Shielding Facility at ORNL and other operating reactors has been used quite extensively. Where the data does not apply directly a semi-empirical approach has been taken, utilizing conservative values of both total and energy absorption coefficients.

D.1.1 Shielding Geometry

As a first approximation to the shielding problem a 5 by 5 lattice of standard BSF fuel elements is considered to be a sphere having an isotropic source spread uniformly over the surface. For the spherical core model, a radius of 30 cm was chosen. This constitutes a compromise between a sphere having the same surface area as the basic BSF core and a sphere whose volume is equal to that of the basic BSF core.

Further assumptions are that the power density is uniform throughout the core volume and that an isotropic medium exists inside as well as outside the

sphere. This simplifies the treatment of the problem.

Plane to Sphere Transformation:

The dose from an isotropic spherical surface source, S_a , is calculated on the basis of the dose from an infinite plane isotropic surface source of the same strength per unit area by the following transformation

$$D_s(r_o, a) = \frac{a}{r_o} \left\{ D_{pl}(r_o - a, \infty) - D_{pl}(r_o + a, \infty) \right\} \quad (1)$$

where $D_s(r_o, a)$ is the dose from a spherical surface source of radius, a , at a distance, r_o , from the center of the sphere. $D_{pl}(r_o - a, \infty)$ is the dose at a distance $(r_o - a)$ from an infinite plane source of the same strength per unit area as the spherical source and $D_{pl}(r_o + a, \infty)$ is similarly defined.

Dose from Spherical Surface Source

Dose from Infinite Plane:

In general, the dose at a distance, z , from an isotropic plane source, S_a , spread uniformly over an infinite plane is

Dose from Plane Surface Source

$$D_{pl}(z, \infty) = \int_{\rho=0}^{\infty} \int_{\theta=0}^{2\pi} S_a \rho d\theta d\rho \frac{e^{-\mu r}}{4\pi r^2} B(r) \quad (2)$$

But

$$r^2 = \rho^2 + z^2$$

and

$$r dr = \rho d\rho$$

Making the above transformation and assuming a linear buildup factor, $B(r) = e^{\mu r}$, Equation (2) becomes upon integrating with respect to θ :

$$D(z, \infty) = \frac{S_a}{2} \int_{r=z}^{\infty} e^{-\mu r} \mu dr$$

Or

$$D(z, \infty) = \frac{S_a}{2} e^{-\mu z}$$

The assumption of a linear build-up factor is a good for thick gamma ray shields and is probably more conservative for neutrons.

In Equation (3), the strength of the isotropic surface source, S_a , is determined from the volume distributed source, S_v , by the following relation

$$S_a = S_v \lambda_c \left(1 - \frac{\lambda_c}{a} \right) \text{ for } \lambda_c < a \quad (4)$$

where λ_c is the appropriate relaxation length in the core material for the particular energy spectrum being considered and "a" is the radius of the spherical core. The assumption, as previously stated, that S_v is uniform throughout the core tends to overestimate the surface source, S_a .

Using Equation (3) for the dose from an infinite plane isotropic source in Equation (1) yields

$$D_s(r_o, a) = \frac{a}{r_o} \frac{S_a}{2} \left\{ e^{-\frac{(r_o-a)}{\lambda}} - e^{-\frac{(r_o-a)}{\lambda} + \frac{2a}{\lambda_c}} \right\} \quad (5)$$

$$D_s(r_o, a) = \frac{a}{r_o} \frac{S_a}{2} \left\{ e^{-\frac{(r_o-a)}{\lambda}} \left[1 - e^{-\frac{2a}{\lambda_c}} \right] \right\} \quad (6)$$

If $\lambda_c \ll 2a$, then Equation (6) can be approximated by

$$D_s(r_o, a) \approx \frac{a}{r_o} \frac{S_a}{2} e^{-\frac{(r_o-a)}{\lambda}} \quad (7)$$

The above approximation is used for the case of a homogeneous isotropic shield.

For the case of a two region shield, (region I followed by region II), it will be of interest to note that the assumption of isotropy in the source at the two-region interface may not be valid.

Let us consider the case of a cosine distribution in intensity of the radiation leaving region I for region II.

With the above assumption Equation (2) becomes

$$D_{pl}(z, \infty) = \int_{\rho=0}^{\infty} \int_{\theta=0}^{2\pi} J_+ \frac{\cos \xi}{\pi} \frac{\rho d\rho d\theta}{r^2} e^{-\mu r} \quad (8)$$

where again a linear build-up factor is employed. J_+ is the radiation current entering region II.

Integrating with respect to θ and making the variable transformation $r dr = \rho d\rho$ gives

$$D_{pl}(z, \infty) = 2 J_+ \int_{r=z}^{\infty} \cos \xi e^{-\mu r} \mu dr$$

but $\cos \xi = z/r$, so

$$D_{pl}(z, \infty) = 2 J_+ \mu z \int_{uz}^{\infty} e^{-\mu r} \mu dr \quad (9)$$

$$D_{pl}(z, \infty) = 2 J_+ \mu z E_2(\mu z) \quad (10)$$

where $E_1(\mu z)$ is defined as (see Reactor Handbook, Vol. I, Table 2.5.1)

$$E_n(X) = \int_X^{\infty} \frac{e^{-p}}{p^n} dp \quad (11)$$

For a shield of several relaxation lengths in thickness Equation (10) becomes

$$D_{pl}(z, \infty) = 2 J_+ e^{-\mu z} \quad (12)$$

From Equation (3), J_+ becomes

$$J_+ = \frac{S_a}{2} e^{-\mu z} \quad (13)$$

where the attenuation factor applied is for region I. With the form of J from (13), Equation (12) yields

$$D_{pl}(\bar{z}_I + \bar{z}_{II}, \infty) = S_a e^{-(\mu_I \bar{z}_I + \mu_{II} \bar{z}_{II})} \quad (14)$$

The dose calculated from Equation (14) is just twice that which would have resulted if the radiation, leaving region I, had been considered isotropic. Since the relaxation lengths that will be applied are considered sufficiently conservative, calculations will be based on isotropic flux distributions in all regions of the shield.

The two region calculations will be based upon the form of Equation (6) which becomes

$$D_s(r_o, a) = \frac{a}{r_o} \frac{S_a}{2} \left\{ e^{-\left(\frac{\bar{z}_I}{I} + \frac{\bar{z}_{II}}{II}\right)} \left[1 - e^{-\frac{2a}{\lambda c}} \right] \right\} \quad (15)$$

where $\bar{z}_I$ and $\bar{z}_{II}$ are the thicknesses of regions I and II respectively and the relaxation lengths are also appropriately chosen.

D.2 SOURCES OF RADIATION

For the purpose of biological shielding calculations, the sources of radiation have been categorized as follows:

1. Prompt fission gammas.
2. Fission product decay gammas.
3. Capture gammas in aluminum, water, graphite and concrete.
4. Fast neutrons.
5. Thermal neutrons.

Except for certain instances in the design of the thermal column, the magnitude of these sources has been determined on the basis of one megawatt operation. The source will also include a fission product build up equivalent to 100 hours operation at maximum power.

D.2.1 Prompt Fission Gammas

The prompt fission gamma source was determined on the basis of two gammas per fission each having an energy of 2.5 Mev. From Section D.1, the equivalent surface source strength of this radiation can now be calculated.

$$S_a = \lambda_c Sv \left(1 - \frac{\lambda_c}{a} \right)$$

The core relaxation length (λ_c) for gamma rays is primarily a composite of the relaxation lengths of aluminum and water. The apportionment of the respective λ 's has been on a weight percentage basis. (see D.2.3).

$$\frac{1}{\lambda_c} = \frac{W_{al}}{\lambda_{al}} + \frac{W_{H_2O}}{\lambda_{H_2O}}$$

$$\frac{1}{\lambda_c} = \frac{0.61}{13} + \frac{0.39}{35}$$

$$\lambda_c \approx 17 \text{ cm}$$

$$S_a = 17 \frac{(3.1 \times 10^{10} \times 10^6 \times 2 \times 2.5 \times 1.6 \times 10^{-6})}{10^5} (1 - 17/30)$$

$$\approx 1.82 \times 10^7 \frac{\text{ergs}}{\text{cm}^2\text{-sec}}$$

It may be well at this point to indicate the procedure used in the selection of the various absorption coefficients. The total absorption coefficient was used in calculating the attenuation factor and the linear build up factor. At the point of interest, the dose rate was determined by using

the calculated transmission together with an energy absorption coefficient. In each case, conservative values have been chosen, i.e., those resulting in a maximum dose rate. Experience has shown that it is not unduly pessimistic in most cases to assume, in the energy range of interest, the minimum value of the total absorption coefficients for all transmission calculations. For transmission calculations, the coefficients were chosen corresponding to a gamma energy of approximately 6 Mev. The energy absorption coefficient in tissue, used to convert to dose rate, was based on a gamma energy of 0.5 Mev ($.032 \text{ cm}^{-1}$). (1,2).

Using the above value for the energy absorption coefficient, the equivalent surface source may be expressed as follows:

$$S_a = 1.82 \times 10^7 \frac{\text{ergs}}{\text{cm}^2\text{-sec}} \approx 2.3 \times 10^7 \frac{\text{Rep}}{\text{hR}}$$

$$S_a = 2.3 \times 10^7 \frac{\text{Rem}}{\text{hR}} \quad (\text{RBE} = 1)$$

D.2.2 Fission Product Gammas

The fission product source was determined by use of the Way-Wigner formula.

$$E = 1.26 \tau_s^{-1.2} \text{ Mev. of gammas/fission-sec.}$$

where E is emitted τ_s seconds after a fission. This gives correct values within a factor of two for times between 10 seconds and 100 days (3,4).

Absorption in the fuel elements and their comparatively short range makes it unnecessary to include energy emitted by β activity.

Upon integration of the above formula, one obtains

$$\delta \text{ power(watts)} = 0.0315 P \left[t^{-.2} - (t_0 + t)^{-.2} \right]$$

where P is the operating power of the reactor (watts), t is the time after shutdown (seconds), and t_0 is the operating time in seconds.

Assuming one megawatt operation for 100 hours, one obtains, as a gamma power, 10 seconds after shutdown:

$$\gamma = 0.0315 (10^6) \left[10^{-0.2} - (10^{-3600 \times 100})^{-0.2} \right] \\ = 17400 \text{ watts}$$

This may be converted to an equivalent surface source in a manner similar to that used for prompt gammas. The resulting dose rate at the core surface is 1.6×10^7 Rem/Hr (RBE = 1). Unfortunately, this figure does not include all activity from short lived fission products existing during reactor operation.

From the integrated Way-Wigner formula, it is evident that 10 seconds after shutdown and for long periods of operation 2% of the total power is from fission products. This figure was arbitrarily increased three times, in view of information indicating 7 per cent of total power while operating is from fission products. Since the 7 per cent includes β and α activity, it was thought that the factor of three would retain the conservative procedure followed in the biological shielding design. Therefore, making similar adjustments in the calculated radiation source, results in a fission product surface source strength during operation of 4.8×10^7 Rem/Hr.

D.2.3 Capture Gammas

The core capture gammas were obtained by calculating the total cross sections for activation of the aluminum and water. The average thermal flux, at one megawatt, was then estimated from BSF data. Neglecting self-shielding, the gammas were obtained from the product of the two.

The core was assumed to be 61% by weight aluminum and 39% water. The

uranium captures have been neglected. The following calculations were based on water captures.

$$\begin{aligned}\sigma_a &= 0.330 \text{ barns/hydrogen atom} \\ \text{Average core density} &= 1.63 \text{ grams/cm}^3 \\ \text{Weight of water in core} &= 1.63 \times 10^5 \times 0.39 = 0.637 \times 10^4 \text{ grams} \\ \text{Total cross section} &= 1410 \text{ cm}^2 \\ \text{Captures/reactor} &= \sum_a^{\text{Reactor}} \phi \text{ Avg. thermal} \\ &= 1410 \times 1.1 \times 10^{13} \\ &= 1.56 \times 10^{16} \text{ captures/reactor-sec.}\end{aligned}$$

Since one gamma (2.23 Mev) is produced for each neutron capture in water, the capture gamma volume source is $5.56 \times 10^5 \text{ ergs/cm}^3\text{-sec.}$ The resulting dose rate at the core surface is $5.1 \times 10^6 \text{ Rem/Hr (RBE = 1).}$

The aluminum core capture gamma source was calculated assuming 2 gammas per capture each with an energy of 5 Mev. These gammas gave an equivalent surface source strength of $7.7 \times 10^6 \text{ Rem/Hr.}$ However, the core is supported by a large grid plate and much of the surrounding structure may also be aluminum. The total equivalent aluminum capture gamma source strength was estimated to be $5 \times 10^7 \text{ Rem/hr.}$

The capture gammas in water outside the core were computed on the basis of an assumed relationship between thermal flux and gamma ray flux; i.e., the variation of gamma ray flux is the same as neutron flux. The following formula was used to determine the capture gamma source strength at the core surface due to captures in the surrounding water (4).

$$\begin{aligned}S_{x=0} &= \frac{I_{x=0}}{2} \left\{ \ln \left| \frac{\mu + \Sigma}{\mu} \right| - e^{-\Sigma L} E, (\mu L) + E, \left[(\mu + \Sigma)L \right] + \right. \\ &\quad \left. + B \frac{\Sigma}{\mu + \Sigma} \left[1 - e^{-(\mu + \Sigma)L} \right] \right\}\end{aligned}$$

$S_{x=0}$ = gamma source (Photons/cm²-sec) at a source plane due to gammas born in water.

Σ = reciprocal of the thermal neutron relaxation length in water (obtained from BSF data) = 0.118 cm⁻¹

μ = the gamma absorption coefficient in water = 0.0286 cm⁻¹

B = buildup factor (assume d linear)

L = thickness of water = 32 inches for bottom of core, 18 inches for side of core.

Σ_a = activation cross section = 0.022 cm⁻¹ for water (H₂)

$I_{x=0}$ = gamma source strength equal to $\Sigma_a \phi_{th}(X)$

$$E_1 = \int_b^{\infty} \frac{e^{-t}}{t} dt$$

The core surface neutron flux was assumed to be a fast flux adjusted from the measured thermal flux with an assumed cadmium ratio (1, 7). The relaxation length for fast neutrons in water was used to determine Σ . The assumption of the low thermal flux at the core surface, would somewhat underestimate the dose rate, but the longer relaxation length used in the region of interest represented a convenient compromise. The gamma source at the core surface facing the main shield (18 inches of water) is calculated below.

$$S = \frac{3.7 \times 10^{12} \times 0.022}{2(.118)} \left\{ \ln \left| \frac{0.1466}{0.0286} \right| - e^{-.118(45.8)} E_1(.0286 \times 45.8) + \right. \\ \left. + E_1(.1466 \times 45.8) + \frac{45.8}{35} \times \frac{0.118}{0.1466} (1 - e^{-.1466 \times 45.8}) \right\}$$

= 9.26 x 10¹¹ gammas/cm²-sec which is approximately equivalent to 4.0 x 10⁶ Rem/Hr (RBE = 1)

In a similar manner, the equivalent surface source strength due to gammas originating in the 32 inches beneath the core was calculated to be 5.2×10^6 Rem/Hr.

The gammas arising from graphite capture were an important contribution to the dose rate from the thermal column and are included in that section.

D.2.4 Fast Neutrons and Concrete Capture Gammas

It was assumed that all the fast neutrons entering the concrete are captured after penetrating a distance equal to the "age displacement" (1). These captures will then give rise to a plane source of gammas. The impinging fast neutron current was determined from BSF core surface measurements and a calculated attenuation through the water.

From D.1

$$D_{\text{sphere}} = \frac{30}{76} \times \frac{3.7 \times 10^{12}}{2} \left[e^{\frac{-46}{8.5}} - e^{\left(\frac{-46}{8.5} - \frac{60}{10} \right)} \right]$$

$$= 3.3 \times 10^9 \text{ neutrons/cm}^2\text{-sec}$$

The fast neutron attenuation length of 8.5 cm in water was a compromise from various reports (4,5,6,7,8,9). The core relaxation length of 10 cm for fast neutrons was determined on a volume fraction basis. Little error is involved in neglecting the uranium content.

$$\lambda_c = \text{Vol \% al} \times \lambda_{\text{al}} + \text{Vol \% water} \times \lambda_{\text{H}_2\text{O}}$$

F.N.

$$\sigma_{\text{removal al}} = 1.31 \text{ barns/atom} \quad (4)$$

$$\sum_{\text{al}} \text{removal} = \frac{(6.02 \times 10^{24})(1.31 \times 10^{-24})}{27} \times 2.7 = .079$$

$$\lambda_{\text{al}} = \frac{1}{\sum} = 12.7 \text{ cm}$$

$$\lambda_{H_2O} = 8.5 \text{ cm}$$

$$\lambda_{FN}^{core} = .37 (12.7) + .63 (8.5) \approx 10 \text{ cm}$$

The calculated neutron current at the concrete surface is four times larger than measured values at the BSF. Since the dose rates from these captures and fast neutrons were small, the above more conservative figure was used.

The concrete capture gammas are all assumed to flow outward from their plane of origin. It can be shown that this is a more conservative procedure than that of considering an exponentially distributed source of capture gammas from the absorption of neutrons in the shield. It has been suggested that the use of an "age displacement" distance in barytes concrete of one foot is satisfactory (1). This distance was used in subsequent calculations.

The fast neutron dose rate has been treated separately. Equivalent core-surface fast neutron dose rates have been attenuated through the interposed shielding. From various reports, adjusted surface value of 3.7×10^{12} fast neutrons $\text{cm}^{-2}\text{-sec}$ has been obtained for one megawatt operation. (1,7) On the basis of 5 fast neutrons/ $\text{cm}^2\text{-sec} = 1 \text{ mrem/Hr}$, this is equivalent to a surface source strength of $7.4 \times 10^8 \text{ Rem/Hr}$ (RBE = 10).

D.2.5 Thermal Neutrons

For biological shielding, the thermal neutrons are rapidly attenuated in the water surrounding the core. The adequate shielding of fast neutrons and gammas reduces the thermal flux to negligible proportions. Any thermal neutrons originating from fast neutrons in the thermal column, are sufficiently attenuated by means of a boron liner.

When the thermal column is in use, graphite effectively replaces much of the water surrounding the core. In this case, the thermal flux is very high. Sufficient reduction still can be obtained with the use of a 3/4 inch boral curtain which may be used to restrict the beam to desired proportions. Such a curtain should give an attenuation on the order of 10^{10} .

D.3 THE PHYSICAL LAYOUT OF THE SHIELDING

The hill-side location for the pool structure as proposed provides for essentially infinite shielding in three horizontal directions from the core. The shielding in the fourth horizontal direction is provided by a thick concrete wall which is pierced by two beam ports and a thermal column. This thick concrete wall extending across the core end of the pool will be referred to as the "main shield" and the remaining portion of the pool as the "pool section". Figure 2.7 shows the general physical layout of the structure.

D.3.1 The Pool Section

The water contained in the pool comprises a significant portion of the shielding material. The pool is 22 feet long, 16 feet wide, and 27 feet deep. This depth allows for 22 feet of water directly over the core.

Horizontal center-line measurements on the BSF at ORNL indicate that 19 feet of water is sufficient to reduce the direct gamma radiation to biological tolerance and that 22 feet of water will further reduce the dose rate to 1/10 tolerance during 1 megawatt operation. The fast and slow neutrons are attenuated even more (7).

Since the direct penetrating gamma radiation from the core in the vertical direction is somewhat less than that in the horizontal direction it is felt that the doses quoted above are conservative from the shielding standpoint.

The dose rate at the surface of the water over the core is strongly dependent upon the flow pattern of water from the core during full power generation. The pattern as described in Appendix A.1.6 will allow the 7-second N^{16} activity induced in the water by the $O^{16}(n,p)N^{16}$ reaction to decay sufficiently before reaching the surface. Even though the cross section for this reaction averaged over the entire fission spectrum is of the order of 0.02 millibarns, (4), considerable activation will result from the cooling water flowing through the high neutron flux region of the core. If this water were allowed to rise directly to the pool surface the 6-Mev gamma arising from the decay of N^{16} would predominate over the direct radiation from the core (10).

D.3.2 Hot Element Storage

Located at the end of the pool opposite the main concrete shield is a trench which will remain filled with water for the storage of activated fuel elements when the main pool section is drained. (See fig. 2.7).

The elements will be set vertically in racks on the floor of the trench so that natural convection of water up through the elements will carry away the decay heat. The storage racks will also provide adequate spacing between fuel elements to maintain a non-critical geometry.

The bottom of the trench has been set at the same level as the floor of the sub-reactor room which is 13 feet below the floor level of the main pool section. With the 2 feet high concrete dam as shown in Figure 2.7 there will be approximately 13 feet of water over the top of the hot elements at all times.

The dose rate over the storage area with the main pool section drained and the trench full of water will be well below tolerance 2 hours after shut-down. Specifically, Moteff's compilations (11) of fission product gamma

attenuation through water indicates that the dose rate with 13 feet of water shield will be 0.2 mr/hr at one day after shutdown. This computation is based on the following assumptions: (1) the core has operated for 1000 hours at a power level of 1 megawatt, (2) the core occupies a small volume and therefore can be treated as a point source, and (3) there is negligible self absorption.

Assumption (2) is pessimistic in this case since the point source was taken to be located at the top of the active source volume of the elements. In addition, the elements will be distributed in rows along the trench which will further reduce the observed dose rate. This effect will not be treated quantitatively in view of the fact that the dose already estimated does not constitute a biological hazard.

Unfortunately the data used did not treat shutdown times shorter than one day. It is of interest to note the dose rate 2 hours after shutdown which is reasonably taken to be a minimum time to unload the elements from the grid plate, place them in the storage trench, and then drain the pool. For this estimate it will be assumed that the source has been decaying as $t^{-0.2}$, where t is the time after shutdown in seconds. This assumption fits the case of long operation. Based on the 0.2 mr/hr at one day after shutdown the dose rate will then be 0.3 mr/hr at 2 hours after shutdown.

D.3.3 The Sub-Reactor Room

Below the main pool section and directly under the reactor core is a shielded room 16 feet long, 12 feet wide, and 9 feet high which houses the machinery and control rod drive mechanisms. The room extends the full width of the pool section and is separated from the hot element storage trench by a barytes concrete wall 5 feet thick.

In order to estimate the gamma dose rate in the sub-reactor room arising from the activated fuel elements in the storage trench the following assumptions were made.

- (1) The Way-Wigner relation for the decay of fission product activity holds.
- (2) The reactor consisting of 20 fuel elements has been operating for a long time (essentially infinite) at one megawatt with a flat power distribution.
- (3) The elements will have decayed for $\frac{1}{2}$ hour. This assumption is based on the fact that entrance to this room will be controlled by the dose arising from the operating core and that it takes $\frac{1}{2}$ hour to place all the elements in the storage trench after shutdown.
- (4) The dose will first be estimated on the basis of one fuel element shielded by 5 feet of barytes concrete, (density = 3.5 gm/cm³.)
- (5) A fuel element will be considered a point source with no self absorption.
- (6) A build-up factor equal to the number of relaxation lengths based on the total absorption coefficient will be applied.
- (7) The total absorption coefficient for 3-mev gammas is taken to be 0.125 cm⁻¹ for barytes concrete and 0.04 cm⁻¹ for water. This energy represents an upper limit for decay gammas from fission products.
- (8) The energy absorption coefficients for tissue will be taken to be a maximum in the energy range of interest,

$$\frac{\mu_e}{\rho} = 0.032 \text{ cm}^2/\text{gm}$$

Upon integration, Equation (1) yields (see Section D.2.2)

$$S_{\gamma} = 0.0315 P \left[t^{-0.2} - (t - t_0)^{-0.2} \right]$$

for $t > 10$ seconds, and

P = power of reactor in watts

t = time after shutdown in seconds

t_0 = operating time in seconds

Assumptions (2) and (3) now give

$$\begin{aligned} S_{\gamma} &= \frac{0.0315 \times 10^6}{20} \times (1800)^{-0.2} \text{ watts/element} \\ &= 3.5 \times 10^2 \text{ watts/element} \\ &= 3.5 \times 10^9 \text{ ergs/sec. (one element)} \end{aligned}$$

Using the above as the source strength and (4), (5), (6), (7) and (8) the dose rate contribution at the face of the concrete inside the sub-reactor room from one element located just opposite the 5 foot wall in the storage trench can now be calculated.

$$\begin{aligned} \text{Dose rate} &= \frac{3.5 \times 10^9 \times 3600 \times 0.032 \times (30.5 \times 5 \times 0.125) \times e^{-19.0}}{4\pi \times (30.5 \times 5)^2 \times 93 \text{ ergs/gm-R}} \\ &= 1.6 \text{ mr/hr (from one element)} \end{aligned}$$

Assuming that a spacing of 1 foot is maintained between elements in storage the observed dose rate in the sub-reactor room will be increased by no more than a factor of 5. This brings the maximum dose rate in the sub-reactor room to biological tolerance from the elements in storage with $\frac{1}{2}$ hour of cooling.

It might be instructive to point out a discrepancy arising between the type of calculation applied here and Moteff's data as used in section D.3.2. The method as used here was applied to the case of 13 feet of water and 2 hours

shutdown time and gave a dose rate of 22 mr/hr from one element. The choice of 3 mev as the average photon energy is extremely conservative for thick shields.

D.3.4 The Main Shield

The main shield was calculated using formulas derived in Section D.1.1 with radiation sources defined in Section D.2. The main shield is composed of barytes concrete in order to minimize space requirements. The composition of the barytes concrete may be taken as:

Portland cement	517 #/cu.yd.
Aggregate	5060 "
Water	316 "

where the aggregate is assumed to be 94% BaSO_4 and 6% SiO_2 by weight. The "cured" concrete has the water content reduced to approximately 78 #/cu.yd., which results in a material of density of approximately 3.5 grams/cm³. From experimental data and theoretical calculations, a fast neutron relaxation length of 8.2 cm was chosen for this concrete. A gamma relaxation length for the barytes was based upon the usual design value for the asymptotic gamma ray relaxation length in ordinary concrete (2.3 grams/cm³) of 15 cm. The relaxation length used (10 cm) has been extrapolated from the accepted value of ordinary concrete by postulating that the relaxation length varies inversely as the mass density (1).

With the above values established, the dose rates may now be calculated. These calculations have involved 2 areas of the shield, i.e.; the radiation transmitted thru the vertical pool retaining wall that transmitted into the sub reactor room.

The geometrical configuration assumed for vertical wall calculations

was a spherical core of 30 cm radius whose center was situated $2\frac{1}{2}$ feet away (in water) from the $7\frac{1}{2}$ foot thick barytes concrete wall. Twenty-two feet of water has been placed above the core to maintain sufficient shielding in the vertical direction (D.3.1).

For the sub reactor room, the interposed shielding consisted of 32 inches of water and 4 feet of barytes concrete. The dose rate in this area is considerably above tolerance and only limited access is allowable. This room is interlocked with the reactor so that a scram occurs upon an unauthorized entrance. However, if, due to an emergency, personnel must work in this area during reactor operation, the dose rate, though high, is not unduly hazardous. This is especially true in view of the fact that some of the following calculations appear to be as much as ten times greater than adjusted experimental data. Radiation streaming affects in this area are negligible.

Vertical Wall Calculations

Radiation sources (See D.2)

Prompt fission gammas	2.3×10^7 Rem/hr
Fission product gammas	4.8×10^7 "
Capture gammas	
Core-water	0.5×10^7 "
Core-aluminum	5.0×10^7 "
Water-side (18")	0.4×10^7 "
Water-bottom (32")	0.5×10^7 "

Fast neutrons

	Core surface	74×10^7 "
Impinging neutron current	Concrete surface (side)	3.3×10^9 $\frac{\text{neutrons}}{\text{cm}^2\text{-sec}}$
	Concrete surface (bottom)	3.4×10^7 "

The radiation has been broken down into three groups for ease of calculation: (a) the fast neutrons, (b) concrete capture gammas, and (c) the remainder, i.e., core and core capture gammas.

$$D_c = \frac{30}{305} \frac{13 \times 10^7}{2} \left[e^{-\frac{46}{35}} - \frac{229}{10} \right] = 0.193 \text{ m Rem/hr}$$

$$D_b = 3.3 \times 10^9 e^{-\frac{(229-30)}{10}} = 7.4 \frac{\text{photon}}{\text{cm}^2\text{-sec}}$$

$$D_a = \frac{30}{305} \frac{74 \times 10^7}{2} \left[e^{-\frac{46}{8.5}} - \frac{229}{8.2} \right] = 0.088 \text{ m Rem/hr}$$

$$= 0.00011 \text{ "}$$

Total dose rate operation @

1 megawatt with fission product $\approx 0.3 \frac{\text{m Rem}}{\text{hr}}$

build up equivalent to 100 hours

at 1 megawatt.

The above indicates that the main shield thickness of 7.5 feet (at the core center line) is adequate for biological shielding.

II Sub Reactor Room Calculations (see also D.3.3)

The sources have been broken down in the same manner as above. Except for the fast neutron current impinging upon the concrete, the radiation sources are essentially the same as above. The water capture source is increased due to increased water thickness under the core. But the rapid attenuation of neutrons in the water outside the core results in only a small increase in this source. Since the source itself is small, the increase has been neglected.

$$D_c = \frac{30}{233} \frac{13 \times 10^7}{2} \left[e^{-\frac{51.4}{35}} - \frac{122}{10} \right] = 4.2 \frac{\text{Rem}}{\text{hr}}$$

$$D_a = \frac{30}{233} \frac{74 \times 10^7}{2} \left[e^{-\frac{81.4}{8.5}} - e^{-\frac{122}{8.2}} \right] = 1.06 \frac{\text{m Rem}}{\text{hr}}$$

$$D_b = 3.4 \times 10^7 e^{-\left(\frac{122-30}{10}\right)} = 3400 \frac{\text{photon}}{\text{cm}^2\text{-sec}}$$

$$= 40.5 \frac{\text{m Rem}}{\text{hr}}$$

$$\text{Total bottom dose} = 4.2 \frac{\text{Rem}}{\text{hr}}$$

As stated previously, adjusted experimental data indicates that this figure may be too large by a factor of ten. If this is true, emergency work may be done in this room during reactor operation.

It would also be of value to know the dose rate in this area after shut down of the reactor. This would give an indication of the waiting time required before entering this room. The other alternative is to unload the core and store the hot elements in the trench, after which this room can be entered immediately. The decay of the fission products has been calculated using the Way-Wigner formula described in Section D.2.2. A fission product build up equivalent to 100 hours of operation at one megawatt has been assumed. At ten seconds after shut down the dose rate at the core surface was calculated to be 1.6×10^7 Rem/hr (see D.2). With this as a source, the dose rate in the sub reactor room is approximately 25 m rem/hr. Similar calculations have been made for periods up to one day.

Time after reactor shut down	Equivalent core surface source strength	Dose in sub reactor room
10 sec.	1.6×10^7 Rem/hr	25 $\frac{\text{m rem}}{\text{hr}}$
10 min.	$.585 \times 10^7$ "	9 "
30 min.	$.43 \times 10^7$ "	7 "
1 hr.	$.34 \times 10^7$ "	5 "
10 hrs.	$.136 \times 10^7$ "	2 "
1 day	$.084 \times 10^7$ "	1.5 "

These dose rates have been calculated on the basis of a gamma relaxation length in barytes of 8 cm. rather than ten. Such a figure is still thought to be conservative due to the softer decay spectrum as compared with that of the operating reactor. Such a procedure has also been followed in the design of the storage trench (see D.3.2).

The high dose rate (during reactor operation) in the sub reactor room is attenuated to well below tolerance ($< 1/10$ tolerance) by the main support wall. Therefore, the area outside this room is safely shielded from the reactor. Also, any scattering path from the sub reactor room over the main shield into this lower area, is sufficient to attenuate the radiation even more. Some problems may be encountered with streaming through the entrance door to the sub reactor room. This can be handled with a concrete baffle or the use of a lead sheathed door. This streaming may not be important in view of the more conservative dose rates from experimental data and due to the conservative geometry, i.e.; transmission calculations based on a minimum amount of shielding (perpendicular) between core and room.

III Geometrical Configuration

The main shield was tapered in both horizontal and vertical planes in order to conserve concrete. Originally, the wetted wall (nearest the reactor) was assumed to be vertical with the outer surface tapered to produce the required shielding. The shape of this wall can be determined from the relation between water and barytes concrete gamma relaxation lengths. For a straight wall, as the radiation emitted from the core approaches the vertical direction, the path through both the water and the concrete increases. Such a wall is larger than necessary and by properly tapering the wall to take account of these changes the amount of concrete used can be reduced. Since the asymptotic

gamma relaxation length in water is approximately $3\frac{1}{2}$ times greater than that of barytes concrete, $3\frac{1}{2}$ feet of water is equivalent to one foot of concrete.

Using this assumption, the following equations can be written:

$$z_a + 3.5 z_b = 27.7$$

$$z_a + z_b = z$$

$$x^2 + y^2 = z^2$$

The 27.7 is the water equivalent in feet of $1\frac{1}{2}$ feet of water and $7\frac{1}{2}$ feet by barytes concrete. Combination of the above equations leads to result:

$$\sqrt{x^2 + y^2} (3.5 - \frac{3.75}{x}) = 27.7$$

The graph of this equation appears in Figure D.3.4a.

It was later decided that the main shield wall should be sloped on both the inner (wetted) side and the exposed (outer) wall. With such a design both sides of the wall are more accessible with the overhead crane. The wetted-wall taper was designed on the basis of a minimum retaining wall thickness of 18 inches. Assuming this base wall to bisect the required 7.5 feet concrete wall (on the horizontal core centerline), the geometric configuration can easily be obtained for the critical case. The critical case or angle is that at which (70° in this case) the shielding consists entirely of water except for the 18 inch concrete "bare" wall. The outside wall was designed graphically since its shape depended largely upon radiation issuing from the thermal column (Figure 6.1). The cross section of the main shield wall in plan view was also designed graphically.

The sides and rear pool retaining walls are all set in the earth to a sufficient depth to provide adequate biological shielding. However, if the pool were to be built largely above the ground level, additional precautions

FIGURE D-3.4a

Shield Geometry

would have to be taken to insure that the dose rates at these walls will be tolerable.

NOMENCLATURE

- a = Radius of spherical core model.
- $D_{pl}(r_0 - a, \infty)$ = Dose at a distance $(r_0 - a)$ from an infinite plane source.
- $D_{pl}(r_0 + a, \infty)$ = Dose at a distance $(r_0 + a)$ from an infinite plane source.
- $D_{pl}(z, \infty)$ = Dose at a distance from infinite plane source, z .
- $D_s(r_0, a)$ = Dose from a spherical surface source of radius distance from the sphere center.
- E = Energy.
- J_+ = Net radiation current, particles $\text{cm}^2\text{-sec}$.
- mr = milliroentgen.
- RBE = Relative biological effectiveness.
- Rem = Roentgen equivalent man.
- Rep = Roentgen equivalent physical
- S_a = Surface source / unit area
- S_v = Volume source / unit volume
- t = time after reactor shutdown, sec.
- t_o = Operating time, sec.
- W = Weight fraction.
- z = Distance from infinite plane source.
- ξ = Angle with the normal to the surface.
- λ = Relaxation length in shield material, cm.
- λ_c = Relaxation length in core material, cm.
- μ = Total absorption coefficient, cm^{-1} .
- $\frac{\mu_e}{\rho}$ = Energy absorption coefficient, $\frac{\text{cm}^2}{\text{gm}}$

- σ = Microscopic cross section, cm^2/atom .
 Σ = Macroscopic cross section, cm^{-1} .
 τ_s = Time after fission, sec.
 ϕ = Radiation flux, particles/ cm^2sec .

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APPENDIX E

WATER PURIFICATION AND FILTER SYSTEM

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E.1 FILTER SYSTEM

Surface dirt and other solid contaminants will be removed from the pool water by continuous circulation of 100 to 250 gpm from the pool surface through a filter and back to the pool. This circuit doubles as the secondary coolant circuit for pressurized water operation or for pool water cooling such as during 1 MW natural convection operation (See Figure A.1.6a). The water always returns to the pool through diffuser nozzles which, during natural convection operation, serve to prevent the hot active water from rising to the top of the pool.

A filter similar to a Dollinger ELS-1380 Staynew Filter should be used. (Manufactured by Dollinger Corporation, 11 Centre Park, Rochester 3, New York). This self-supported unit of corrosion resistant steel has 6" flange connections and swing-bolt type head removal. Its weight is about 800 lbs, and overall dimensions are 32" diameter x 0.85" high. The WC-5-1 woven cotton filtering medium is in the shape of reinforced accordian-shaped cylinders providing 138 ft² of filter area, and efficiently removes particles down to 30-40 microns. The pressure drop when "clean" is 2 psi at 180 gpm. The filters can be easily changed when the pressure drop becomes prohibitive (about 10 psi).

E.2 PURIFICATION SYSTEM

The important reasons for maintaining the pool water as pure as possible are as follows:

- 1) Ions which act as corrosive catalysts toward aluminum and its alloys are kept down; hence, corrosion rates and scaling are minimized.
- 2) Water activity during reactor operation and equipment maintenance is greatly reduced.

- 3) Impurities in water which reduce reactor reactivity are kept at a minimum.

Mixed bed deionization was chosen over conventional deionization or bypass distillation because it is more economical and has worked satisfactorily at the BSF for more than a year. The original water treatment of the BSF consisted of adding 60 ppm of sodium chromate to the pool water. Operating experience indicated that in order to minimize corrosion a demineralizer should be added and the chromate addition dropped. Corrosion, visibility, and radiation background were improved by this change. It should be mentioned here that the pool walls will be "Amercoated" to prevent leaching of minerals from the concrete. The white coating will also improve visibility and facilitate decontamination of the pool walls should the pool become contaminated.

The design of a resin bed is influenced by the amount of water in the system, expected accumulation rate of ionized solids due to corrosion and other contamination, desired frequency of bed regeneration and replacement, type and capacity of resin used, and throughput rate. Since some of these factors are practically impossible to estimate, the BSF demineralizer has been chosen for this system. It was chosen over four other demineralizers for having the largest capacity per dollar installed cost (5) and will handle the BSF system which is almost twice as large as the PETR pool.

E.2.1 General Description

The demineralizer circuit is operated as a 6 gpm continuous bypass of the filter effluent through a booster pump. The circuit consists of the demineralizer and a small filter to remove resin carryover. The demineralizer can be connected to the primary coolant loop for pressurized water operation or the secondary coolant loop by proper valving.

The demineralizer is an Elgin Ultra Deionizer (manufactured by Elgin Softener Corp., Elgin, Illinois) with a rated capacity of 68,000 grains of CaCO_3 . The bed consists of 3 ft³ of Amberlite IR-120 (strong acid cation resin) mixed with 4.5 ft³ of Amberlite IRA-401 (strong base anion resin). It will require 21 lbs of caustic soda and 21 lbs of 66° Baumé sulphuric acid for regeneration. The demineralizer tank, regenerant tank, water quality indicator, and associated piping, valves, and flow meters require a space approximately 3' x 4' x 8' high. Maximum operating pressure, temperature, and flow rate are 100 psi, 140 F, and 18 gpm respectively.

For removal of resin carryover a small filter similar to a Cuno "Micro-clean" filter model 3AX1-2278 (manufactured by Cuno Engineering Corp., Meriden, Connecticut) should be installed. This unit is approximately 9" diameter x 21" high, made of bronze, and employs a cluster of three 25 micron filter elements 2 3/4" dia x 9 3/4" long. Clean pressure drop is 2 psi at 22 gpm.

The BSF pool has been emptied several times into White Oak Creek for access and has entailed no health hazards. For construction at a college or university it would be desirable to install a storage tank while the pool is empty since discontinuous operation of the demineralizer tends to lower the effluent quality. For construction at ORNL sufficient demineralized water can be obtained from the LITR to fill the pool.

Resistivity of the effluent is measured by an indicating instrument supplied with the demineralizer; however, much more information about the unit operation could be obtained from a resistivity indicator and a pH indicator in the bed inlet water stream. Resistivity and pH must be measured in the flowing stream as samples are easily contaminated by their containers and CO_2 absorption.

from the air. Inlet resistivity would indicate the quality of the pool water and combined with the effluent resistivity would indicate the efficiency of the bed. Commercial industrial conductivity recorders using nickel cells are reliable up to 1 megohm-cm, above which platinum metal electrodes are recommended. For bulk water resistivities below 1 megohm-cm, the pH record of the resin column influent is a better index of resin bed depletion.

E.2.2 Pressure Drop

The demineralizer circuit is shown in Appendix A, Figure 1.7a. The pressure drop around the loop was calculated in a manner similar to that of Appendix A. Pressure drop through the resin bed is shown in Figure E.2.2a. as a function of flow rate in gpm.

Pressure drop through the Elgin demineralizer unit was calculated for 70°F water at 18 gpm.

Resin Bed	3.6 psi
Demineralizer tank	0.4 psi
Piping, valves, etc of unit	7.1 psi
Filter (maximum)	<u>5.0 psi</u>
	16.1 psi

Design Conditions and Calculations

Water temperature - 70°F (for max ΔP)

$$\rho = 62.3 \text{ lb/ft}^3$$

$$\mu = 0.982 \text{ centipoise or } 0.00066 \text{ lb/ft-sec}$$

$$\omega = 18 \text{ gpm}$$

$$u = \frac{18}{60(0.006) 7.48} = 6.7 \text{ ft/sec in 1" pipe}$$

Resin Bed - 1.8 ft^2 cross-sectional area

Amberlite IR-120 = 1.63 psi*

Amberlite IRA-140 - Total $\Delta P = \frac{1.90 \text{ psi}}{3.53 \text{ psi}}$ *

*(mixed resin data not available)

Demineralizer Tank

Entrance loss

$$= K_e \frac{\rho u^2}{144 (2g)} = \frac{(1) 62.3 (6.7)^2}{144 (64.4)} = 0.3 \text{ psi}$$

Exit loss

$$P = K_c \frac{\rho u^2}{144 (2g)} = \frac{(.4) 62.3 (6.7)^2}{144 (64.4)} = 0.12 \text{ psi}$$

$$\text{Total } \Delta p = 0.42 \text{ psi}$$

Piping and Fittings

Fittings

Equivalent ft of 1" pipe (8)

5 elbows	14.0
2 run of std tee	3.6
5 std tee through side outlet	30.0
1 gate valve, open	0.6
1 gate valve, 3/4 open	3.5
1 flow meter (estimated)	20.0
1 conductivity cell (estimated)	10.0

81.7

$$Re = \frac{D u \rho}{\mu} = \frac{1 (6.7) (62.3)}{12 (.00066)} = 5.3 \times 10^4$$

$$f = 0.006 (7)$$

$$\Delta P = \frac{2 f L u^2 \rho}{g D (144)} = \frac{(.006) (2) (81.7) (6.7)^2 (62.3)}{32.2 (1/12) (144)} = 7.1 \text{ psi}$$

FIGURE E-2.2a

E.3 NOMENCLATURE

D	= Diameter, ft
K _c	= Contraction coefficient
K _e	= Expansion coefficient
L	= Length, ft
P	= Pressure, psi
u	= Velocity, ft/sec
R _e	= Reynolds number
f	= Friction factor
g	= Acceleration of gravity - 32.2 ft/sec ²
ρ	= Density, lb/ft ³
ω	= Flow rate, gal/min
Δ	= An increment
μ	= Viscosity

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APPENDIX F

COST ESTIMATE

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APPENDIX F

F.1 General

In general, wherever it was possible to do so, the costs indicated are based on quotations from recognized suppliers of the material indicated. In the case of construction costs the figures are based on recent experience with similar structures.

F.2 REACTOR

The following costs are based either on a quotation or on ORNL cost records. The figures used include installation and freight costs.

Structure (Supports, grid, plate, pressure shell, etc.)	\$17,600
Fuel Elements (50 elements at \$200 per element)	10,000
Control rods (4 at \$600)	2,400
Control rod drives	7,100
Instrumentation	<u>20,000</u>
Total	\$57,100

F.3 COOLING SYSTEM (installed costs)

Pumps	\$4,200
Valves and Piping	6,000
Heat Exchanger	5,000
Cooling Tower	12,000
Filler	500
Ion Exchanger equipment	2,800
Instrumentation	<u>2,000</u>
Total	\$33,200

F.4 BUILDING

<u>Item</u>	<u>Amount</u>	<u>Unit Cost</u>	<u>Item Cost</u>
Foundation and floor slab reinforced concrete	195 cy	\$100/cy	\$19,500
Excavation (earth	200 cy	6/cy	1,200
(rock	270 cy	60/cy	16,200
Structural steel building	2,560 ft ²	12/A ²	30,700
Misc. structural mat'l.	5 tons	800/ton	4,000
Lean-to structure	1,020 ft ²	15/ft ²	15,300
2 Ton monorail hoist			4,500
Total			\$91,700

F.5 POOL STRUCTURE

High density concrete	170 cy	\$190/cy	\$32,300
Reinforced concrete	150 cy	100/cy	15,000
Excavation (40% earth	370 cy	6/cy	2,200
(60% rock	550	60/cy	33,000
Grating	200 ft ²	250/ft ²	500
Misc. Structural steel	2 tons	800/ton	1,600
Amer coating	2600 ft ²	751/ft ²	2,000
Total			\$86,600

F.6 SUMMARY

Reactor	\$57,100
Cooling System	33,200
Pool Structure	86,600
Building	91,700
Misc. (site preparation utilities, etc)	25,000
	<u>\$293,600</u>
Contingency	26,400
	<u>\$320,000</u>
Engineering	30,000
	<u>\$350,000</u>

Note that these summary figures do not include such experimental facilities as the thermal column, beam ports, loops, etc. They do, however, include the necessary provisions in the structure for their inclusion. Note also that the costs of the pool structure and the building contain approximately \$45,000 allowance for rock excavation which would not be needed if some site other than ORNL were contemplated.

F.7 EXPERIMENTAL EQUIPMENT

Because of the somewhat tenuous nature of the costs of the experimental auxiliaries, we have chosen to set them out separately. The final cost figures for these items will depend strongly on the degree of refinement of the final design and on the choice of instrumentation.

Thermal column	\$30,000
Beam ports	3,000
Instrumentation (Health physics, counters, etc.)	12,000
Fluid fuel "loop"	20,000
Fixed fuel "loops"	<u>15,000</u>
Total	\$80,000

APPENDIX G

EXPERIMENTAL LOOPS

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G.1 CIRCULATING FUEL LOOP

In order that experiments may be performed on a reactor with circulating fuel characteristics, a loop was designed for insertion into the central portion of the PETR core. This was thought to be the most economical and practical method of obtaining circulating fuel characteristics.

G.1.1 Mechanical Arrangement

The basic loop is fabricated of 347 stainless steel canned in aluminum. This choice was made for reasons of compatibility with the core material and corrosion resistance to the fuel solution. The entire device is monitored for fuel solution leakage by circulating helium through the loop between the aluminum and steel walls and then passing this helium through a radiation detector and monitoring this gas stream.

The loop is described in Dwg. G.1.1a and consists of a "U" tube 2.500" diameter in the core region, a heat exchanger of double wall construction, a canned rotor pump and an expansion tank. The heat exchanger is constructed of 55 bi-metallic externally ribbed tubes composed of a 0.2500 I.D. 347 SS inside an aluminum tubes. This allows a helium leak detection stream to pass down each tube. The maximum corrosion rate of properly pre-treated 347 stainless steel is 0.008 inches per year so that with average operating cycles the loop should last a minimum of two and one-half years before needing replacement. The heat exchanger cooling water is brought from the primary cooling system up through two 3.00 inch O.D. aluminum tubes to the heat exchanger than from the heat exchanger it discharges directly to the pool.

G.1.2 Heat Transfer

The heat exchanger is of shell and tube type construction with one pass counter flow. It is composed of fifty-five 250" I.D. by 0.350" O.D. bi-metallic

Circulating Fuel Loop.

FIGURE G-1.1a

tubes located on one half inch centers in a 5 by 11 square array. The total heat exchanger length is 36 inches. This gives a heat transfer area of 10.8 ft^2 based on the tube I.D.

At design flow conditions of 45 gpm of the fuel, and 50 gpm of the water coolant the overall heat transfer coefficient of the tubes is 383 BTU/hr ft^2 $^{\circ}\text{F}$. Thus at maximum operating conditions of water temperature into the exchanger at 100°F and fuel entrance temperature at 200°F the heat exchanger will remove a maximum of 100 kilowatts.

G.1.3 Pressure Drop

The pressure drop through the loop was calculated in the usual way (similar to Appendix A) and may be represented by the equation:

$$\text{Head Loss} = 0.0345 (\text{gpm})^{1.85} \text{ ft}$$

This equation is shown plotted in Figure G.1.3a along with the pressure characteristics of a suitable pump (Chempump type CF with 5-3/4 inch impeller). Thus the design condition is at 45 gpm or a velocity of approximately 2.95 ft/sec in the core section, with a pump speed of 3450 gpm. To obtain different flow rates the pump speed may be varied and the loop flow rate calibrated for different pump speeds.

G.1.4 Physics

The circulating fuel chosen was of five weight per cent fully enriched uranium in the form of uranyl sulfate in aqueous solution. With this fuel in the loop the nuclear parameters in the core are:

FIGURE G-1.3a

<u>Gm/cm³</u>	<u>Material</u>	<u>Macroscopic Absorption Cross Section cm⁻¹</u>
0.0035	U-238	0.00002
0.0465	U-235	0.07150
0.0190	O	0.00000
0.0063	S	0.00006
0.5200	H ₂ O	0.01150
0.3220	Al	0.00000
0.00016	He	0.00000
0.00035	C	0.00000
0.0074	Mn	0.00010
0.0035	Si	0.00001
0.0630	Cr	0.00185
0.0390	Ni	0.00164
0.2360	Fe	0.00564

The critical mass and flux pattern for the reactor with the loop inserted are shown in Appendix D.

G.2 SOLID FUEL ELEMENT LOOP

For purposes of testing the hydraulic and nuclear characteristics of solid fuel elements of various types and designs, a fuel element testing section similar to that shown in Figure G.2.1a may be used. This device consists basically of a thin 347 stainless steel shell located in the center of the reactor core. Cooling water is provided through two 3 inch diameter steel tubes that connect from the cooling water inlets to the top of the shell. The cooling water passes downward through the test section and out through orifices in the bottom of the test section.

Figure 0-2.1a
Solid Fuel Element Loop

Cooling water is metered by two standard, square-edged, A.S.M.E. orifices, one located in each coolant supply tube. The solid fuel element loop as proposed would be furnished in either of two sizes. Loop number one would place four fuel elements and would allow testing of elements of similar size to the PETR elements one at a time. Loop number two would replace nine elements in the core and would allow for testing PETR size elements four at a time.

Total critical mass calculations and flux patterns have been calculated for loop number one with one PETR element, and four loop number two with four PETR elements. These results are shown in Appendix D.

APPENDIX H

CONTROLS AND INSTRUMENTATION

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APPENDIX H

CONTROLS AND INSTRUMENTATION

H.1 GENERAL

The control and safety circuits for the PETR are similar to those used on the BSF. These circuits have been operated reliably for several years. The only difference in this system stems from the rods being hydraulically-driven rather than motor-driven. The system consists of two neutron-level safety channels, one counting rate channel, one period channel, and one linear channel with which a servo control is associated (Figs. H.3.1a, H.4.1a, H.4.1b). Due to the various core configurations that are possible, one regulating rod and three safety rods can be installed in any one of nine possible positions to provide the most effective control. The "fail safe" philosophy is employed through the system.

Two PCP chambers, two compensated ion chambers, and a fission chamber are mounted on an aluminum frame which rests on the grid plate, being positioned by empty fuel element holes. The chambers are mounted in a manner such that they may be placed at several different distances from the core. The cables run along the wall to the top of the pool through a flexitallic metal tube. This tube waterproofs the cable bundle, and, being flexible, allows the chamber frame to be lifted to the top of the pool for direct maintenance and positioning of the chambers. The cables pass through a plug which fits in a groove at the edge of the grid plate allowing the pressure dome to sit flush on the grid plate during pressurized operation.

FIGURE H-1.1a PETR CHAMBER FRAME

H.2 HYDRAULIC ROD DRIVES

H.2.1 Hydraulic Circuit

The hydraulic circuit for the rod drive is shown in Figure H.2.1. It is intended that the rod-drive be entirely built of commercially available components. The positioning of the rod is done by a 4 inch double-acting hydraulic cylinder having a double extended rod. With the exception of the span circuit, the system is entirely conventional.

Power to move the system is supplied through a four-way three-position solenoid-operated valve from a vane-type hydraulic pump circuit. The four-way valve has a double solenoid controls so that energizing the solenoid offsets the position of the spool. De-energizing the solenoid allows the springs to bring the spool of the valve back to the neutral position at which point the ports are all blocked and movement of the piston is prevented. Scram action of the circuit is obtained by means of hydraulic pressure provided by a gas-loaded accumulator which operates on by-pass from the main system. Pressure is applied to two hydraulically controlled check valves which prevent the opening of the scram circuit until power is lost. On loss of power to the scram solenoid, the spool of a valve controlling the hydraulic checks is spring shifted to open these control lines. At this point a 1-1/4 inch line is opened from the bottom of the piston to the oil reservoir and a 1-1/4 inch line is opened from the accumulator to the top of the piston, driving the piston down and inserting the poison control rod.

This circuitry is repeated for each control rod. There are two motors provided, one motor driving two control rod pumps. The entire complement of components for the system is mounted in one package on a common 75-gallon reservoir. The only hydraulic connections required between the hydraulic

package and the piston will be the 1-1/4 inch line.

H.2.2 Mechanical Arrangement

The piston rod of the drive system is hollow. The stem of a toe rod passes down through the center of the hollow piston rod and is attached to the rod by a single nut at the bottom exposed end of the rod. In order to change control rods, this nut is removed, the cap is screwed on over the end of the rod and the rod is then pulled out through the top of the core. After the control rod has been pulled from this position, the entire cylinder and seal assembly can be removed by plugging the top end of the spring guide cylinder with a temporary plug. If it is desired to move the drive to one of the alternative positions, a blank flange can then be installed to cover the opening which is left upon removal of the cylinder. In order to provide for possible loss of hydraulic pressure, a spring is included within each rod position which acts to drive the rod downward inserting the poison. The bottom end of the hydraulic cylinder also carries a selsyn drive with a spring-weave rewind drum. The cable from this drum is attached to the bottom end of the piston rod in order to drive the selsyn. This then allows remote indication of the position of the rod as desired. This arrangement is shown in Figure H.2.2a.

H.3 SYSTEM

The philosophy behind the design of the safety system was to create a system with simplicity and reliability. The safety system is designed to fail-safe upon either power or equipment failure. The system as shown in Figure H.3.a resembles the system used successfully in the bulk shielding facility reactor at Oak Ridge National Laboratories for the past five years.

FIGURE H-3.1a

The safety chambers comprise two parallel-plate, boron-coated ionization chambers about 3 inches in diameter which provide a current proportional to the neutron plus gamma flux level and one compensated ion chamber, also about 3 inches in diameter, the current from which is proportional to the neutron flux. The primary safety system provides a minimum period scram, a maximum power scram, and a manual and health physics monitor scram. The parallel-plate ionization chambers are connected to the primary side of a small DC amplifier. The output of this amplifier is fed to the sigma amplifier and thence to the sigma bus. The solenoid scram valves are controlled by a separate amplifier. As the current from the parallel plate ionization chamber increases with the combined flux, the voltage on the sigma bus increases to a present maximum value and opens a relay shutting off the power to the solenoid amplifier, scrambling the reactor.

The output of the compensated chamber is fed to a period amplifier where voltage output is adversely proportional to the period of the reactor power rise. This output voltage is fed to the sigma bus where scram may occur in a manner similar to that described for the parallel plate ionization chamber. The safety rods are raised by a hydraulic cylinder that is valved so that scram occurs whenever the power supply to the solenoid amplifier is off. The reactor may be scrambled manually by loss of control rod driving pressure or from a health physics monitor operation by grounding the sigma bus. This then reduces the sigma bus voltage below a preset value and opens a relay in the solenoid amplify system, shutting off the amplifier. This will cause a scram to occur as before.

H.4 CONTROL AND INSTRUMENTATION

H.4.1 Control System

The control system as designed is provided with a hydraulic type servo to control the rod position for constant power operation with slightly varying water temperatures. The reactor power level is monitored by a fission chamber, operating a count-rate channel for low powers, and a compensated ion chamber for high power levels.

The count-rate circuit is shown in Figure H.4.1a and consists of a fission chamber, a linear preamplifier, and a linear amplifier feeding a scaler and a log count-rate recorder. When reactor operation is at high powers, above 10,000 counts per minute, where the count-rate circuit will no longer discriminate, the fission chamber high voltage power supply may be turned off.

The compensated ion chamber circuit as shown in Figure H.4.1b consists of: A chamber sensitive, primarily, to the neutron flux; a micro-ammeter with a remote range preselector switch; a Brown flux recorder; and a solenoid amplifier.

The reactor system may be operated either manually or from servo control by selecting the proper switch position on the solenoid amplifier output. The range of this chamber is such that, when combined with the range of the fission chamber, the reactor may be operated at any power from zero to one megawatt.

H.4.2 Instrumentation

The instrumentation provided is as shown in Figures H.3.1a, H.4.1a and H.4.1b and consists of both visual and recording instruments. These instruments are: In the count-rate channel - a scaler, a log count-rate meter, and a log

FIGURE H-4.1A

FIGURE H-4.1b

count-rate recorder; in the flux circuit - a Brown flux recorder; and in the safety circuit - a log recorder, a period recorder and a period meter.

This instrumentation is the minimum for successful operation of the reactor. Instrumentation for use with engineering and physics experiments has not been included in this section.

APPENDIX I

THERMAL COLUMN

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I. THERMAL COLUMN

I.1.1 Introduction and Description

The following criteria and specifications for design of the thermal column will be assumed without proof in this section.

- 1) Column will fit in a concrete wall 7 1/2 feet thick, this thickness being specified for purposes of biological shielding from the reactor at one megawatt operation.
- 2) Cross section of the column will be four feet by four feet.
- 3) A removable section of graphite canned in aluminum will fit in the 18 inch water space between the reactor face and a one-half inch aluminum water-barrier. The plug will be designed so that a minimum path exists for neutrons in water in their travel from the reactor through the thermal column. A fit loose enough to facilitate removal of plug with ease is also necessary.
- 4) Materials inside the column will be restricted to graphite and aluminum.
- 5) Design will be such that no crack or interface exists in the column in a direction 54 inches long perpendicular to the reactor face. This requires one offset in the column.
- 6) The shielding requirements will be determined for one megawatt operation at infinite time. The maximum allowable dosage of radiation will be 7.5 milli-rem per hour with the thermal column closed. Somewhat higher than tolerance dose rate will necessarily be permitted in some regions around the experimental facilities, when in use, but it is intended that these regions be sufficiently well defined to constitute no serious biological hazard.

- 7) A boron liner around the graphite will be installed to stop thermal neutrons. See Figure 6.1.
- 8) A lead liner around the outside of the column will be used in conjunction with the main concrete shield wall to reduce to tolerance the gamma flux leaving the sides of the thermal column.
- 9) The graphite shall fill all space in the column not occupied by the lead curtain and shall be flush with the outside surface of the main shield.
- 10) For conversion from flux to dose, the minimum total absorption coefficient and maximum energy absorption coefficients within the range of energies expected are used. This is a conservative procedure which appears to have wide application elsewhere in design of reactor shield. (1,7).
- 11) In all cases where doubt as to the actual numbers to use exists, values were chosen which were known to be conservative, i.e., which overestimated the radiation dose appearing at the outside of the main concrete shield.
- 12) Isotropic distribution of gammas and neutron fluxes in all cases.
- 13) Design criteria for lead curtain
 - a) At 100 watts operation, with thermal column opened, the gamma dose streaming from the hole in the graphite shall not be greater than 6 rem per hour.
 - b) At one megawatt operation, with column closed, and after infinite time of operation at this power, the dose rate at the outside column surface from capture gammas in graphite shall be approximately equal the dose from core gammas.

Core gammas are defined as the sum of capture gammas in aluminum and water, prompt core gammas, and fission product decay gammas.

There is an advantage for the dose from core gammas at this surface to be less than the dose from capture gammas when it is desirable to provide a region where the flux is composed of neutrons rather than gammas.

- c) The total weight of lead in curtain and face shields shall be a minimum.
- 14) A minimum of 10^4 thermal neutrons/cm²-sec is required for a flux plot (5).

The general philosophy of approach to shielding the thermal column is to simplify the normally monumental task of handling calculations requiring a knowledge of both energy and spectral distribution of flux throughout the thermal column by assuming the energy of all particles or photons to be the maximum possible for each given source. Isotropic distribution is assumed and the flux relaxed through the media present using minimum relaxation lengths in all cases.

For the case of radiation from the core, the usual infinite plane source was converted to a spherical source, thus introducing a geometrical attenuation factor (4). For sources closer to the outside shield face, the use of a spherical source appears to be unjustified and an infinite plane source was used instead.

The practice of assuming the energy of all particles to be a maximum in the source is not expected to seriously overestimate the shield thicknesses required. It is the higher energy particles in each case which get through

and do most of the radiation damage, necessitating a weighting of the damage effects in favor of the higher energies.

A consideration of the kinds and thicknesses of shielding material required will necessitate the examination of the contribution from the following sources of radiation.

- 1) Prompt gammas from the core.
- 2) Delayed gammas from fission products.
- 3) Capture gammas from core materials.
- 4) Capture gammas from graphite.
- 5) Capture gammas from thermal column lining material.
- 6) Fast neutrons from core regions.
- 7) Slow neutrons from core region.

Thermal neutron leakage at the surface of the Bulk Shielding Facility operated at one megawatt is reported to be 1.1×10^{13} neutrons/cm²-sec and fast leakage at this same location as 3.7×10^{12} neutrons/cm²-sec (6). For a cold clean core at this same power level and position, the measured gamma dose was 8×10^7 rem/hr. These measurements were taken as maxima for PETR at the same specified conditions and location. These are basic assumptions as used in the calculation of shielding thicknesses for PETR.

I.2 SHIELDING

I.2.1 Lead Door and Curtain Thicknesses for Capture and Core Gammas at One Megawatt Operation.

It is assumed that the thermal flux follows the form:

$$\phi = \phi_0 e^{-x/L'} \quad \text{-----} \quad (1)$$

where L' is the apparent diffusion length in the finite graphite column, cm. (See Sample Calculations.)

x is the distance, cm, from the reactor face into the thermal column, cm.

ϕ_0 is the initial flux at the reactor face, neutrons/cm²-sec.

ϕ is the flux at "x", neutrons/cm² sec.

and L' can be found from the relation: (23)

$$1/L'^2 = 1/L^2 + 2 (\pi^2/a^2) \quad (2)$$

from the actual diffusion length, L, to the apparent diffusion length, L'.

Equation (1) will not accurately represent the slow neutron flux within two diffusion lengths from the reactor because of the effect of fast neutrons, nor will it apply close to the back of the column because of end effects. However, the representation is considered adequate since the worst variation near the reactor core will present a total contribution of radiation dose at the back of the graphite column which is relatively insignificant.

I.2.2 Lead Door and Curtain in Thicknesses for Capture and Core Gammas at 100 Watts Operation

Infinite Time of Operation at 1 Megawatt, and With Lead Door Open

The slow neutron flux at the surface of the reactor was estimated to be a linear function of the power level, e.g., a 10^{+4} factor decrease in power level indicating a 10^{+4} decrease in thermal neutrons, giving a flux at the reactor outside face of about 10^9 . A decrease through the graphite column by a factor of 10^5 maximum is anticipated if the flux falls off exponentially as $e^{-x/L'}$. This results in a minimum flux of about 10^4 neutrons/cm²-sec, the specified minimum to allow for adequate irradiation in the 4" by 4" graphite hole of specimens for flux plots (5).

In the calculations following, a lead window thickness of four inches was arbitrarily assumed.

The dosage at the outside face of the column due to captures in the graphite was calculated before. The decreased thermal flux both initially thermal and fast was attenuated exponentially according to $e^{-x/L'}$ to determine the thermal neutron flux throughout the column. Gamma production was estimated by multiplying the flux by the graphite absorption cross section. The resulting volume sources were converted to surface sources and attenuated exponentially throughout the varying path lengths. A conservative estimate of no exponential attenuation throughout the length of the graphite hole was assumed. These calculations gave a 1.6 contribution of rem per hour from capture gammas in graphite.

Core gammas also contribute to the dose rate at the graphite hole. A calculation from the Wigner-Way formula⁽²⁾ gave an estimate of the gamma activity one hour after the reactor is shut down following operation at one megawatt for a long time. This contribution was 3.16×10^6 rem per hour at the reactor face. The core gammas present as a result of operation at 100 watts for one hour proved to be negligible. Assuming no self shielding, a dose rate of 3.7 rem per hour for a case of 4 inches of lead in the curtain was estimated.

The total dose from gamma radiation streaming through the 4 x 4 inch hole is conservatively estimated at 5.2 rem per hour for operation at 100 watts. The maximum allowable dose rate was arbitrarily set at 6 rem per hour.

Thus, it appears that the earlier assumption of 4 inches of lead in the curtain was a reasonable one. A further effort to minimize this thickness is unjustified in view of the small saving of material to be effected.

As seen previously, a 10 inch door with no curtain gave a dose rate of only 1.7 milli-rem per hour while a 10 inch door with a 5 inch curtain gave a dose rate of 1.0 milli-rem per hour. Therefore, a 10 inch door and 4 inch curtain

were finally selected as meeting the requirements of shielding for both conditions of operation at one megawatt and 100 watts.

I.2.3 Lead Liner Thickness for Graphite Capture Gammas

The lead liner was arbitrarily assumed to have the following dimensions.

The first 54 inches is 2 inch thick. The next 28 inches is 4 inches thick. The last 8 inches is increased to a total of 6 inches by the proper placement of lead shielding around the column perimeter. See Figure 6.1.

In order to prove that the dose rate around the door will be within limits, this design was examined for the worst possible condition (one megawatt operation) and with the assumption that all graphite capture gammas produced at a point are scattered into the lead liner traversing it at a minimum path length of two, four, or six inches. The gamma photon is then assumed to be scattered out of the concrete shield along the minimum path length through it. This latter step would necessarily involve scattering into a 90° angle and a resultant loss of energy (See Sample Calculations.)

Starting at the reactor face and working toward the outside of the thermal column, one divides the column into sections approximately 2 diffusion lengths (46 cm) in thickness. Assuming the flux varies through the column as $e^{-x/L'}$, a flux distribution is obtained. The flux is averaged over the front and back surface of each slab to obtain an average neutron flux as representative of the given slab. The fast neutron flux is treated as before. Having averaged slow neutron flux values, an average gamma production value is obtained for each slab. This is changed to a surface source by the appropriate geometric transformation. Each slab is assumed to be emanating, from the plane nearer the column door, the above mentioned average gamma flux.

A summation of the various contributions, after attenuating the original sources through the varying lead and barytes concrete thicknesses, gives a value of 70 milli-rem per hour. This figure assumes no loss of gamma energy due to scattering and assumes a minimum flight path. Actually, this high dose rate will never be experienced, for in scattering through 90° in order to arrive at the minimum path length through the main shield wall, the 6 mev gamma loses 5.5 mev or 90 per cent of its original energy if it reaches the outside as a result of one scattering (See Sample Calculations).

If the 6 mev gamma reaches the outside as a result of 2 scatterings in its minimum path flight, 88 per cent of its original energy is lost.

It appears safe to conclude that the dose rate of 70 milli-rem per hour will be at least cut in half as a result of either a decrease in original gamma energy as a result of scattering several times, or the gamma seeking a path length greater than the minimum distance through the shield. If not one, then the other procedure must necessarily follow in its attenuation effect. This dose rate does not appear to be excessive for an intermittently used experimental facility.

The effects of gammas streaming from the crack between the door and main shield wall can be neglected due to the initial gamma ray energy loss sustained in scattering into position for escaping through the crack and the small solid angle available for entry into the region outside of the main concrete shield. It is also clear that the passage way between the door and shield will have the effect of columnating the gamma flux. Normally the area affected will be subject to infrequent use and can be purposely avoided.

In order to minimize the cost for concrete in the main shield, the minimum path length in concrete for attenuation of each gamma source plane to

tolerance dose rate was found using the minimum appropriate path length in lead for the given geometry. As each path length was determined, a circle of path length radius was placed on a scaled drawing of thermal column orientated in main shield with extreme end of source plane as focal point.

Thus it was determined that the main shield wall could be tapered 60° from a point starting 4.0 feet from the outer periphery of the thermal column. (See Figure 6.1).

I.2.4 Fast Neutrons

The fast neutron flux at the reactor face was estimated to be 3.7×10^{12} neutrons/cm²-sec (1-7). Also, it was conservatively assumed that none of the fast neutrons were captured. Attenuating this current through 8 feet of graphite, the radiation dose at the shield surface was found to be appreciably lower than one-tenth tolerance without any shielding other than graphite.

I.2.5 Thermal Neutrons (boral front and liner)

Maximum shielding will be needed at the outer surface of the column where there is no concrete to aid in the slow neutron attenuation. If the physical situation is over simplified by assuming that no attenuation of thermal neutrons is accomplished throughout the length of the column, there will be approximately 10^{13} thermal neutrons/cm²-sec escaping from the outside face of the graphite at a power level of 1 megawatt. But a 3/4 inch boral shield at the face of the reactor will attenuate this by a factor of 10^{10} —sufficient to make the dose rate of thermal neutrons negligible compared to the dosage from sources of capture gammas nearer the back of the column.

Graphite having an absorption cross section of 3.2 mb will be used.

Also, one 6 mev gamma will be produced per capture of a thermal neutron.

Slow neutrons arising from slowing down of the fast flux also contribute to gamma production throughout the column. All fast neutrons were assumed to be thermalized at a distance t/λ from the core face. Conservatively, this distance was increased from 39 to 46 centimeters to coincide with number (2) source. At this point, the sum of the two sources was considered to continue its distribution throughout the column according to equation (1). An adjusted fast flux value of 3.7×10^{12} fast neutrons/cm²-sec was used⁽¹⁻⁷⁾.

Having the thermal neutron flux throughout the column, volume sources at a series of planes approximately two diffusion lengths apart were calculated. This volume source was changed to a surface source according to:

$$\sum_a \phi \lambda = S_v \lambda = S_a \text{ - - - - - (3)}$$

where λ is the attenuation length of gammas in graphite, 24 cm. Considering each of these sources to be infinite planes, arithmetic flux averages were taken between the several planes and each fictitious source attenuated exponentially. Contributions from the several planes were then added to give the dose rate at the surface of the graphite.

As a convenience, "core gammas" was defined as the combination of prompt gammas from the core, delayed gammas from fission products, and capture gammas from aluminum and water. The "lead curtain" is the lead layer between the aluminum water barrier and graphite column. The outer lead shield is referred to as the "lead door".

It is evident from a consideration of item (13) of the design criteria that a compromise is indicated in arriving at an optimum distribution of lead between the curtain and door of the column.

Initially, assuming a five inch lead curtain, a dose rate of 180 rem

per hour was calculated as the contribution of capture gammas in the graphite appearing at the reactor face for one megawatt operation. To attenuate the core gammas to the same order of magnitude (106 rem/hr) would require the removal of all lead in the curtain. Arbitrarily specifying a 10 inch thick lead door for the cases of zero and 5 inches lead in the curtain, the following dose rates were calculated immediately outside the 10 inch door.

Curtain (inch)	Door (inch)	Total thickness (inch)	mr/hr
5	10.0	15.0	1.0
0	10.0	10.0	1.7

A justification of the 1/4 inch boral liner was made on the basis of examining its efficiency at its weakest point. This occurs when the full flux available at the outside face of the column is directed across the 1/4 inch Boral liner at right angles to the liner. This flux has been conservatively estimated previously to be 5.4×10^8 thermal neutrons/cm²-sec. It is clear that the attenuation will bring the dose rate for thermal neutrons within tolerable limits. Glasstone gives a value of 2000 thermal neutrons/cm²-sec for the tolerance dosage⁴.

I.3 FABRICATION AND INSTALLATION

I.3.1 Introduction

The overall design is shown in Figure (6.1). Many of the constructional details have been omitted because of time limitations. Insofar as possible an effort has been made to keep the construction as simple as possible using conventional materials and shapes.

I.3.2 A Method of Construction

Before pouring the heavy density concrete, place in position a form fabricated from heavy timbers and cross braced inside so as to retain the outside dimensions of the thermal column, $-0 + 1/8"$. A masonite liner on the outside of the form would serve to finish the inside concrete surface.

It would be desirable to place a series of anchor bolts in the inside form for the main concrete shield wall so as to allow them to become embedded in the wall to a depth of 6 inches after pouring concrete. These bolts would ring the inside hole, giving access to the column, on 6 inch centers and would each be 18 inches from the hole edge. Their purpose would later be to draw the 7 feet square aluminum water barrier up against a gasket set in the face of the wall thus making the seal water tight.

The "I" beams, supporting the roller mechanism for the thermal column door, would be placed in position prior to pouring concrete. Also place in position the channel in which the door bottom roller bearings go. Place in position the cooling mat coils around the inside perimeter of the thermal column.

After the concrete has been poured, set, and the forms stripped, place in position the aluminum water barrier and secure same.

The bottom and side liners, lead and boral, are placed and the graphite laid in, in proper position, the fit of each piece having previously been checked by laying up outside of the hole.

After the graphite up to the interface is laid, the lead and boral coming up to this interface, from the inside, can be placed on top of the graphite. The remainder of the graphite is then laid and the remainder of the top shield placed in position.

The channel in which the door top roller bearings fit is welded in position and the doors swung on their bearings. Each bearing support has provision for vertical adjustment so as to line up the plug opening with the removable 4 inch by 4 inch graphite plug.

The plug door is a 7" x 7" x 6" piece of lead in a channel iron framework. The hinge consists of a one inch rod appropriately bent after being swung into the 1" round holes drilled into two 2-1/2 inch iron cubes. These cubes are welded, as shown in diagram, to a 6 inch channel. The ends of the one inch rod are then welded to the channel iron frame of the plug door. A spring, as shown, may be attached to keep door in normally closed position.

The thermal column door is in two sections, the sections rolling in opposite directions to expose the thermal column face. Each section is a mirror image of the other and is fabricated by welding together 10 x 2-5/8" channel and "Z" iron with 2 x 2 inch angle braces. The use of the Z iron accomplishes an offset in the door sections intersection preventing gamma streaming.

As can be seen from the diagram, a 5 by 5 inch hole giving access to the column is present when the closure is made. The plug door acts as the biological shield for this hole.

Figure (I.3.2)

Vertical aluminum angles on 12 inch centers, with welded construction, will give the necessary strength to the 1/2 inch aluminum water barrier in resisting lateral stresses. One spacing of 18 inches is made between the reactor and thermal column so that the angles will not result in a water gap between the graphite plug and aluminum water barrier and will aid in holding the plug in place.

I.4.d Cooling

As a result of absorption of gammas and fast neutrons in the lead and boral liner when the reactor is operating at high power, it will be necessary to cool the first 54 inches of the column by a natural convection cooling mat surrounding the perimeter of the column.

As envisioned, the cooling mat will consist of a series of 1/2 inch copper "coils" on 6 inch centers set 2 inches inside the concrete surface. These "coils" will be cooled by natural circulation of the pool water

I.5 SAMPLE CALCULATIONS

See nomenclature for definitions of all symbols.

Calculation of radiation dose contribution, D, from capture gammas in graphite can, appearing at the face of thermal column. Reactor is operating at 1 megawatt.

$$\begin{aligned}
 S_a &= \Sigma_a \phi \lambda \quad \Sigma_a \text{ is } 0.268 \times 10^{-3} \text{ cm}^{-1} \\
 \frac{\text{Gamma}}{\text{cm}^2\text{-sec}} &= \frac{1}{\text{cm}} \frac{\text{neut}}{\text{cm}^2\text{-sec}} \frac{\text{gamma}}{\text{neut}} \text{ cm} \\
 &= 0.268 \times 10^{-3} \times 24 \phi \\
 &= 3.22 \times 10^{-3} \phi \text{ gamma/cm}^2\text{-sec}
 \end{aligned}$$

At the graphite can surface (Source number 2),

$$\begin{aligned}
 \text{Rem/hr} &= \frac{\text{Rem-cm}^2\text{-sec}}{\text{gamma-hr}} \frac{\text{gamma}}{\text{cm}^2\text{-sec}} \\
 &= 11.9 \times 10^{-6} \times 3.22 \times 10^{-3} \phi \\
 &= 3.83 \times 10^{-8} \phi \text{ Rem/hr} \\
 &= 3.83 \times 10^{-8} \times 8.1 \times 10^{12} \\
 &= 3.11 \times 10^5 \text{ Rem/hr}
 \end{aligned}$$

$$\begin{aligned}
 D &= \frac{S_a}{2} e^{-x/\lambda_{\text{Pb}}} e^{-z/\lambda_{\text{G}}} \\
 &= \frac{3.11 \times 10^5}{2} e^{-10.15/2.12} e^{-218/24}
 \end{aligned}$$

for attenuation through 10.15 centimeters of lead curtain and
216.3 centimeters of graphite.

$$D = \underline{\underline{145 \text{ mr/hr}}}$$

Calculation of apparent diffusion length in graphite.

$$1/L'^2 = \gamma_{11}^2 - 2 \pi^2/a^2$$

where

$$\gamma_{11} = 1/L'$$

$$1/(50)^2 = 1/L'^2 - 2 (3.14)^2/(122)^2$$

$$L' = \underline{\underline{24 \text{ cm}}}$$

Fig. I-5.0a Schematic diagram showing division of thermal column into sources for purpose of calculating thermal flux distribution.

Slab Source	Average thermal neutron flux neutrons/cm ² -sec
1 - 2	0.81×10^{13}
2 - 3	3.6×10^{12}
3 - 4	10.9×10^{11}
4 - 5	1.6×10^{11}
5 - 6	2.3×10^{10}
6 - 7	3.4×10^9
7 - 8	0.54×10^9

Contribution from core gammas after 100 hours of operation at 1 megawatt at graphite face as follows.

See elsewhere (D.2) for the calculation of dosage present at face of the core as a result of capture gammas from water and aluminum prompt fission gammas and fission product gammas. Addition of these three items gives 13×10^7 rem/hr as the dosage rate from core gammas at the reactor face. Assume this to be an infinite plane source, convert to a spherical source and attenuate exponentially.

$$\begin{aligned}
 D &= r/r_o \quad S_a/2 \quad e^{-x/\lambda} \\
 &= \frac{30}{229 \cancel{+} 46 \cancel{+} 30} \quad \frac{13 \times 10^7}{2} \quad e^{-10.15/2.12 - 265/24} \\
 &\approx \underline{\underline{0.81 \text{ rem/hr}}}
 \end{aligned}$$

Contribution from graphite capture gammas in graphite can present at column face when operating at 100 watts.

$$\begin{aligned}
 D &= \frac{3.83}{2} \times 10^{-8} \times 8.1 \times 10^{12} \times 10^{-4} \\
 &\quad \times e^{-10.15/2.12 - 218/24} \\
 &\approx \underline{\underline{0.014 \text{ mr/hr}}}
 \end{aligned}$$

Calculation of graphite capture gamma energy loss due to a scattering collision deflecting gamma photon at an angle of 90° to original path.

$$\begin{aligned}
 E_1 &= \frac{E_o}{1 + \frac{E_o}{.51} (1 - \cos \theta)} \\
 &= \frac{6}{1 + \frac{6}{.51} (1 - 0)} \\
 &= \frac{6}{1 + 12}
 \end{aligned}$$

$$= .46 \text{ Mev}$$

$$\text{Energy loss} \approx 6.5 = 5.5 \text{ Mev}$$

Percent of original energy lost

$$= \frac{5.5}{6} \times 100 \approx 92 \%$$

General calculations showing optimization for thermal column lead door and curtain.

Arbitrarily assign a thickness of 10 inches to lead door.

Consider case for 1 megawatt operation:

The dose rate at the outside face of the graphite from graphite capture gammas is 180 rem per hour, the method indicated in section I-2.1.

At section 8 (see Fig. I-5.0), the core gamma contribution, D, is:

$$D = \frac{13 \times 10^7}{2} \times \frac{30}{229 + 46 + 30} e^{-\frac{12.7}{2.12} - \frac{263}{24}}$$

$$\approx 0.27 \text{ rem per hour}$$

with 5" Pb curtain

Now, what thickness Pb curtain will give core gamma dose rate of same order magnitude as graphite capture gamma dose rate?

Assume no Pb curtain.

$$D = 0.27 e^{-(12.7/2.12)}$$

$$= 106 \text{ rem per hour}$$

Therefore, a zero thickness Pb curtain is justified by this criterion. With the 5" Pb curtain and 10" door, what is the dose rate?

$$D = (180 + 0.27) e^{-\left(\frac{25.4}{2.12}\right)}$$

$$= 180 \times 10^{-5.22}$$

$$\approx 1.0 \text{ milli-rem per hour}$$

With no Pb curtain and 10" lead door, what is the dose rate, D, from core and capture gammas?

$$106.0 + 180 = 286$$

$$D = 286 e^{-25.4/2.12}$$

$$D = 1.7 \text{ milli-rem per hour}$$

Case I

5" curtain

10" door

15" total

1.0 milli-rem/hour

Case II

0" curtain

10" door

10" total

1.7 milli-rem/hour

At 100 watts: Specify a 4" Pb curtain.

The capture gamma dose rate is 1.6 rem per hour at the outside graphite face as indicated in I-2.2.

What is core gamma dose at the outside graphite face?

$$D = \frac{3.16 \times 10^6}{2} \times \frac{30}{305} e^{-10.15/2.12} - 140/24$$

$$= 3.65 \text{ rem/hour}$$

$$\text{Total dose} = 1.6 + 3.6$$

$$\approx 5.2 \text{ rem per hour}$$

Considering 5.2 rem per hour not excessive, a 4 inch Pb curtain is chosen. The lead door thickness has previously been chosen as 10 inches.

I-6.0 NOMENCLATURE FOR APPENDIX I

a = side of a square cross section of thermal column, cm.

E_0 = original gamma photon energy, mev.

E_x = energy of deflected gamma photon, mev.

D = dose rate, rem/hr.

L = actual diffusion length in graphite column, cm.

L^1 = apparent diffusion length infinite column, cm.

r = radius of spherical source, cm.

r_o = distance from center of spherical source to detector, cm.

S_a = surface source strength, gammas/cm²-sec.

S = volume source strength, gammas/cm³-sec.

Σ = cross section, cm⁻¹.

λ = attenuation length, cm.

t = Fermi age, cm².

$\gamma_{11} = 1/L^1$, ft⁻¹.

Subscripts

G = graphite

I-7.0 REFERENCES FOR APPENDIX I

- 1) "ORR Shielding Studies I - Basic Design Information", by N. F. Lansing, ORNL-CF-54-7-95.
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- 3) "The Elements of Nuclear Reactor Theory", by Samuel Glasstone and Milton C. Edlund, D. Van Nostrand Co., New York.
- 4) "Principles of Nuclear Reactor Engineering", by Samuel Glasstone, D. Van Nostrand Co., Inc., New York.
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- 6) "Attenuation by Water of Radiations from a Swimming Pool Type Reactor", by G. M. Maienschein, et al, ORNL-CF-1891.
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