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The swimming pool nuclear reactor.

Radcliffe, John B.

Monterey, California: U.S. Naval Postgraduate School

THE SWIMMING POOL NUCLEAR REACTOR

JOHN B. RADCLIFFE Jr.

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THE SWIMMING POOL NUCLEAR REACTOR

JOHN B. RADCLIFFE, Jr.

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ITS CHARACTERISTICS, USES, AND RELATIVE MERITS

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by

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Submitted in partial fulfillment
of the requirements
for the degree of
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IN
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MASTER OF SCIENCE
IN
PHYSICS
FROM THE
United States Naval Post-Graduate School

PREFACE

The subject of the low power research reactor was initially conceived as a topic for a feasibility study designed to probe the merits of having a research reactor facility at the United States Naval Post Graduate School. Survey of available information on reactors quickly led to the conclusion that time and security factors would not permit a thorough analysis of all research types available. Instead it was decided that a sound contribution could be made toward such a study by selecting the prototype of one unclassified research reactor of considerable merit, collecting and digesting the scattered information available, and analyzing it in the light of the basic considerations which must be fundamental to any feasibility study. It is believed that this project will aid anyone conducting such a study at a future date, and will serve as a reliable reference guide, should this type of reactor be selected as the desired facility.

The writer would like to express his appreciation to Drs. Austin R. Frey and E. C. Crittenden Jr. of the United States Naval Post Graduate School. He is indebted to both for guidance in the selection of the topic, and for advice and encouragement in pursuing the research.

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CHAPTER I

INTRODUCTION

1. Summary.

In order to arrive at any conclusions concerning the relative merits of low power research reactors, it is necessary to become acquainted with their uses and limitations. In doing so, the inescapable conclusion is that even though a research reactor project is to be approached with caution, the tremendous possibilities presented to nearly all departments of a graduate level engineering institution make the research reactor a facility of inestimable value.

In selecting a reactor type it is necessary to consider at least three basic factors: versatility, safety, and cost. It is virtually impossible to make quantitative comparisons concerning details of specific reactor types. Each one is designed for a specific purpose, and with changing designs the variables involved have wide ranges of latitude. The only reasonable comparison, then is a qualitative one on the basis of the three factors cited above. Although the limited scope of this work precludes a thorough comparison of even unclassified types, the conclusion to be drawn is that the swimming pool reactor stands very high on all three considerations - possessing extreme versatility of use, a high degree of safety, and a low cost. These characteristics of the reactor are set forth in detail in Chapter III, together with discussions of reactor use and problems of reactor research.

In presenting a general description of the reactor, it

was decided to consider all of the reactor variables in the light of two operating power levels, 100 KW and 1 megawatt, with appropriate comments concerning lower and intermediate levels of operation. The two levels specified were selected as being representative of the range of operation to which this type of reactor is suited. Most of the descriptive material in Chapters II through IV is applicable to both designs; but where the details diverge, both aspects are included for basis of comparison.

A few of the technical considerations have been included in Chapter IV, both as items of interest concerning certain fundamental types of calculation (such as that for criticality), and also to clarify certain detailed aspects of the general information in Chapter II.

2. Tabulation of Characteristics and Costs.

This section represents what the writer believes to be the most efficient means of presenting a summary of a variety of details - a tabulation of the characteristics and costs. The uses do not lend themselves well to brief summary, hence they are discussed in Chapter III.

TABLE I

SUMMARY OF REACTOR CHARACTERISTICS

Power Level: 100 KW

Type: Thermal neutron, heterogeneous, light water, research

Fuel: Enriched uranium ($>90\%$ U^{235}) in Al alloy sandwiches containing 30-40 gms of U^{235} in each of 4-5 parallel plates, 3x(.1)x24 in. spaced $\frac{1}{2}$ in. apart in each fuel element.

Critical Mass: 3 kg U^{235} in 16 fuel elements (core 12x12x24 in.)

3.5 kg including beamholes and burnup

2.4 kg with Be-O reflector (core 9x12x24 in.)

For 1 megawatt: (calculated)

2.2 kg clean core (water reflector)

3.0 kg operating core (water reflector)

1.2 kg with Be-O reflector

Reflector: 3 in. Be-O on 4 sides of lattice; (or light water)

Moderator: light water

Coolant: light water (convective flow)

For 1 megawatt: light water, forced flow (2 ft/sec), from 1250 gal/min, 50 ft head pump.

Shielding: 16½ ft of water above, 3⅛ ft water below, 3 ft concrete on sides and bottom.

For 1 megawatt: 21 ft water above, 4 ft below, 8 ft concrete on sides (centerline)

Flux: Thermal neutrons: 10^{12} n/cm^2/sec

Epi-thermal: 3×10^{12} n/cm^2/sec

1 megawatt: 8×10^{12} (average thermal flux in core)

1.2×10^{13} (experimental flux at reflector surface)

Control: 2 B-Pb shim-safety rods (mixture of Pb and boral in oval Al can 1x2½x26 in), one control rod, same dimensions, actuated by gravity, and electro-magnetic control (~5% each in Δk).

1 megawatt: 3 B-Pb safety rods (Δk~6% each)

1 control rod

Power Density: 2 watts/cm^3 (light water reflector) in active lattice

1 megawatt: 20 watts/cm^3

Heat Transfer from Fuel Plates: $1\frac{1}{2}$ watts/cm^2 = 5000 BTU/hr ft^2 (light water reflector)

1 megawatt: 84,000 BTU/hr-ft^2

Temperature Coefficient: - .0075 %/°F

Aluminum to Water Ratio: 0.3

TABLE II

SUMMARY OF REACTOR COSTS

For 100 KW (convective cooling design)[1]

Cost of Fuel Elements: $120 ea., 20 elements: (exclusive of uranium)	$ 2,400.00
Reactor Assembly: labor, overhead and materials (motors, magnets, grid, superstructure, etc)	28,000.00
Electronic Circuits: labor, overhead, materials (chambers, circuits, recording inst., etc.)	28,000.00
Servo-Automatic Control	2,600.00
Total Reactor and Controls:	$61,000.00
Desirable Auxiliary Equipment:	
Be-O reflectors (30 elements)	17,000.00
Spares (Chambers and electrical equipment)	9,000.00
Health Physics Instruments	7,500.00
Total:	$33,500.00
Estimated Cost of Housing Structure: (14x18x22 ft pool in bay 28x28x30 ft., with 2500 sq. ft. lab space)	125,000.00
Grand Total :	$219,500.00
Additional Estimated Costs for 1 Megawatt Design[2]:	
Pumps and Heat Exchanger	15,000.00
Additional instruments and spares	5,500.00
Miscellaneous (including somewhat larger housing structure)	31,000.00
Total:	$271,000.00

1. Based on exact costs in ORNL records of 1950

2. Estimated from data of AECD 3557. (See detailed breakdown, pp 39-40)

CHAPTER II

THE SWIMMING POOL REACTOR

1. Historical Background.

The Bulk Shielding Facility, popularly known as the Swimming Pool Reactor, was first constructed at the Oak Ridge National Laboratory. It was devised initially to be used for experiments to aid development of improved reactor shields. Having gone critical on December 17, 1950, the Reactor was put in operation shortly thereafter, and the details of its operation were declassified early in 1952. In three years of operation, it has been observed that the reactor has a substantial variety of potential research uses and possesses certain features of safety and economy which make it one of the foremost possibilities for adaptation by institutions interested in nuclear research.

2. General Description.

The Swimming Pool is a thermal neutron, heterogeneous reactor, generating heat on the order of several hundred KW (with convective cooling) to 1 megawatt (with forced cooling). Light water is employed as moderator and coolant. The water plus the concrete walls of the pool serve as the shield material. Fuel consists of enriched uranium contained in removable sandwich type Al-Ur alloy plates (>90% U^{235}), with a critical mass of approximately 3 kg. Be-O or light water can be used as reflectors. Only 2 to 3 B-Pb safety rods are required, and one control rod of similar construction. Control is aided slightly by a negative temperature coefficient of -.0075%°F. The maximum available

slow neutron flux at 100KW is 10^{12} n/cm^2/sec. Flux increases to ~1.2×10^{13} for 1 megawatt operation with Be-O reflector. The active lattice is suspended from an instrument bridge in an aluminum framework which is submerged in a large pool of water sufficiently deep to serve as a shield (~20 ft) The bridge rests on wheels fitted to rails along the sides of the pool, so the reactor can be moved the length of the pool along the center line. The bridge also carries a serve control mechanism by which a movable carriage is brought up or down, placing instruments at any point in the pool. A three dimensional view of the reactor, pool, and controls is shown in Figure 1.

3. Fuel Core Design and Burnup.

The reactor core consists of twelve to eighteen fuel elements held in an aluminum grid. Each element is an assembly of 5 parallel plates, approximately 3 in. wide, 0.1 in. thick, and 24 in. long, spaced $\frac{1}{2}$ in. apart. Each plate contains 30 to 40 gms of U^{235} in the form of Ur-Al alloy encased in 2S Aluminum alloy sandwiches. The assembly is hot rolled into a solid plate, preventing escape of fission products. The design of the fuel element was based on the requirement for dissipating on the order of 100 kw of heat by convective cooling. It is primarily this condition which necessitates a 5 plate per element design with a consequent aluminum-to-water ratio of 0.3. Experiments have revealed that the fuel element plates retail all of the fission product yield. Moreover, since the fissioning of 50 mg of U^{235} results in one megawatt hour of energy, several years of low power operation are provided before fuel elements need be repro-

cessed. Higher power operation shortens this period, and increases the rate of accumulation of fission-product poisons. However, the problem is relatively minor under any circumstances. Facilities are available through the AEC for reprocessing of the spent fuel elements. They are made at ORNL at a cost of $120 each (minus cost of uranium). A detail sketch of the fuel element is shown in Figure 2. Modifications of this design are easily made. For example, three special fuel assemblies are constructed in which a pair of fuel plates are more closely spaced toward each side, allowing space in the center for entry of two safety rods and the control rod. (See Figure 3). In addition, four fractional assemblies (20, 40, 60 and 80% normal fuel content) can be constructed in order to hold the excess "k" to as small a value as possible, and to permit geometrical symmetry for varying loading patterns.

The fuel elements are arranged vertically in an aluminum grid, held in place by a conical end box welded to the bottom of each element. The grid holes are 5 in. deep, providing sufficient stability for the elements without further support. In the Oak Ridge model, the grid contains 54 holes, a large enough number to provide for variations in the fuel loading pattern, and for dummy elements containing reflector material or specimens to be irradiated. A photo of the grid and the partially loaded active lattice is shown in Figure 4.

4. Moderator.

The pool water performs a triple function - that of moder-

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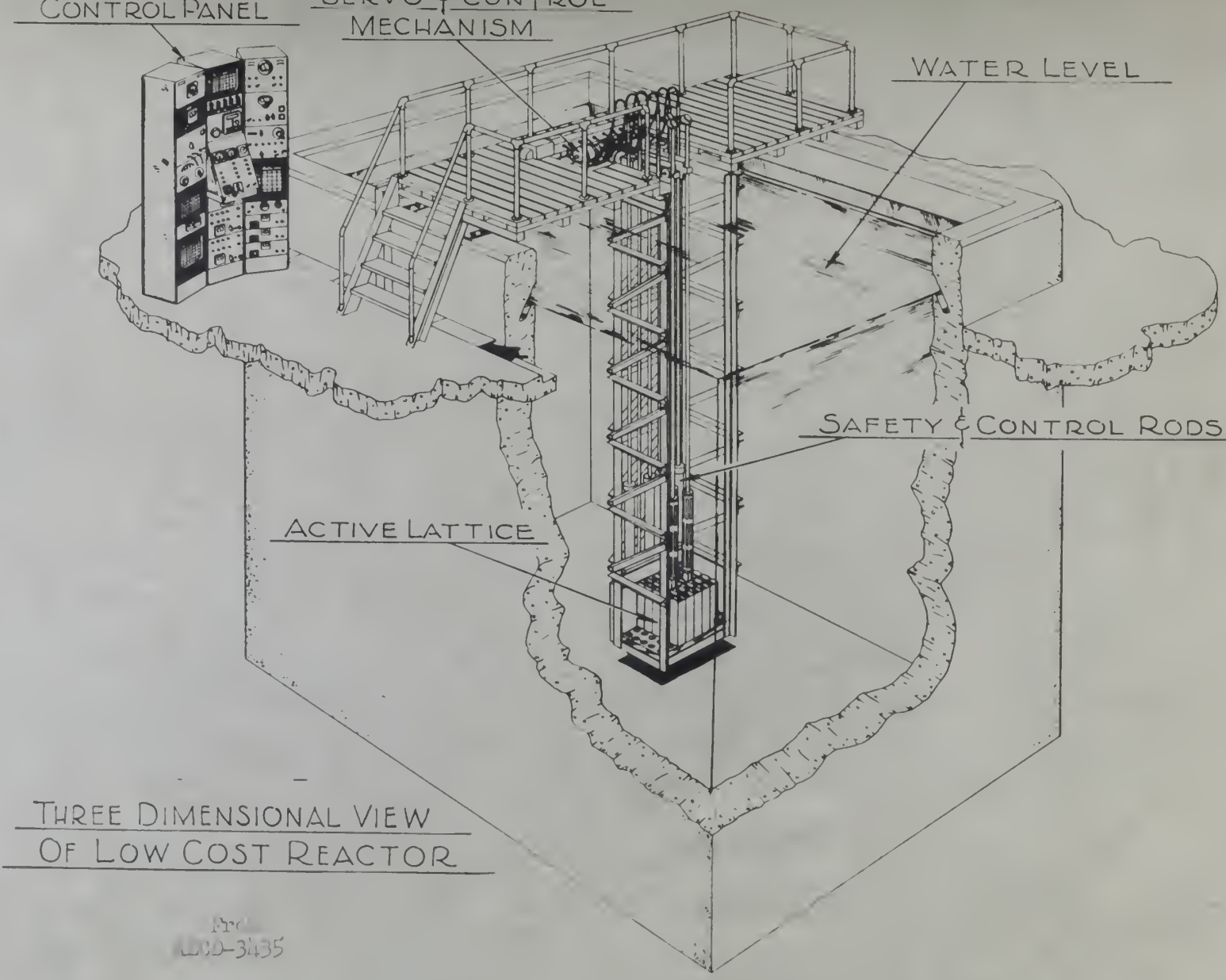

THREE DIMENSIONAL VIEW
OF LOW COST REACTOR

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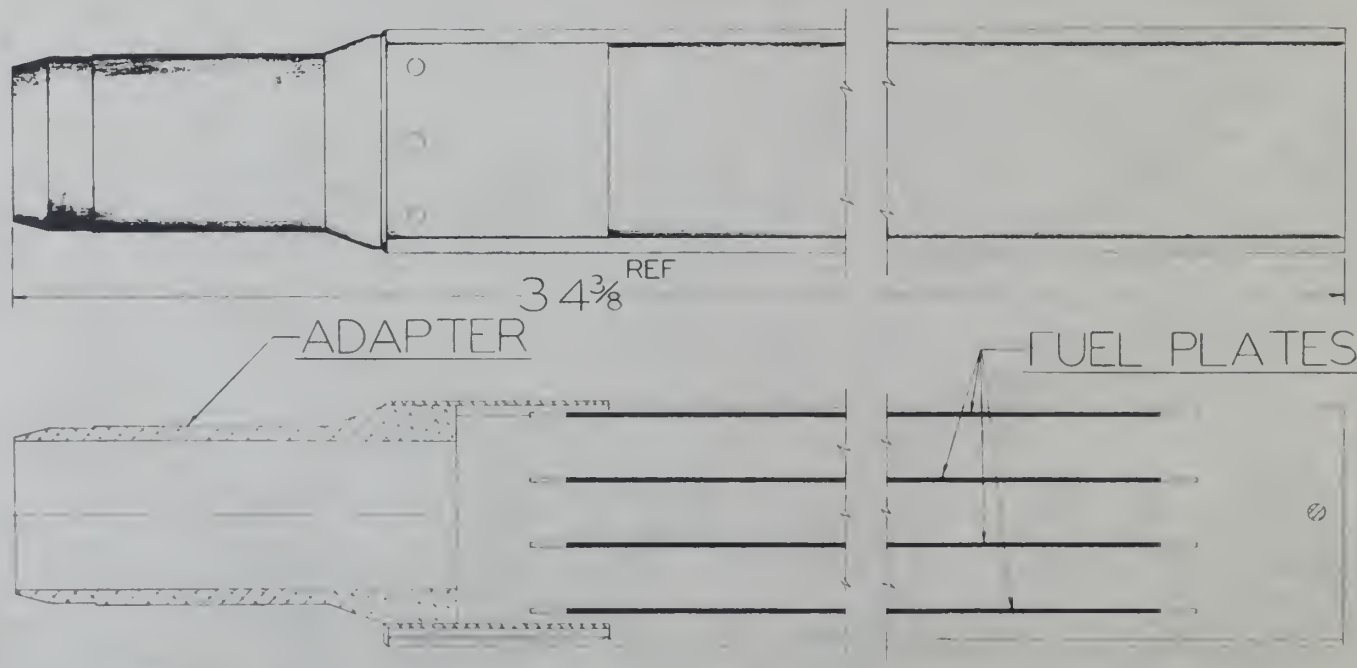

FUEL ASSEMBLY
LOW COST REACTOR

FIG. 11

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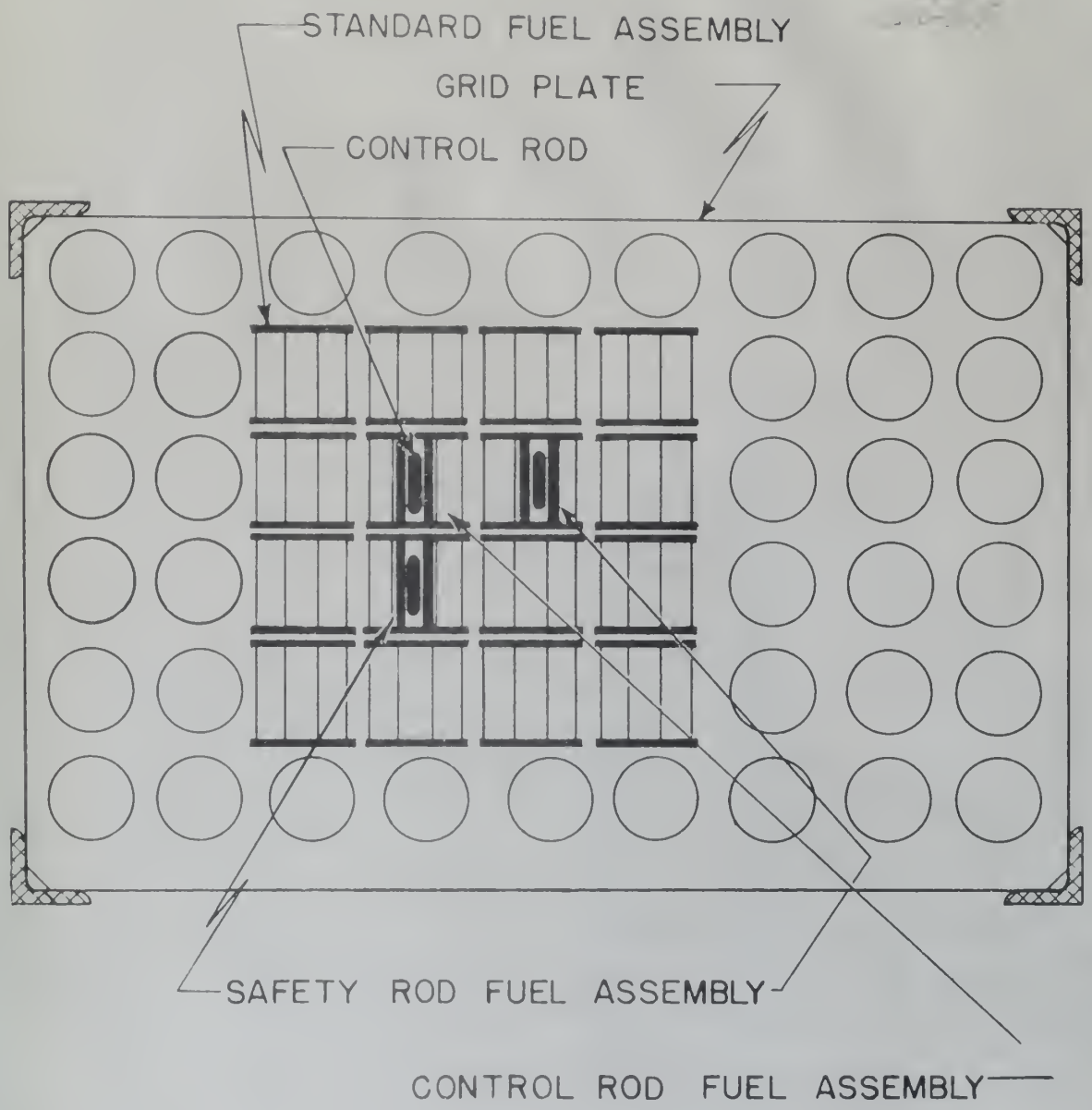

ONE SUGGESTED LOADING PATTERN
LOW COST REACTOR

REACTOR LATTICE STRUCTURE

PHOTO 7415

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Fig. IV

ator, coolant, and primary shield (also as a potential reflector if a better reflector is not provided). The high scattering cross section (164 barns for .025 ev neutrons) and low atomic weight makes water the best inexpensive substance available as a moderator in thermal reactors. The effectiveness of a moderator is indicated by its moderating ratio, $\frac{\sigma_s \xi}{\sigma_c}$, which for water has an exceptionally high value of 250. Moreover, at normal power operation (100 kw), there is relatively little activation of the water, and the slow rate of diffusion of water to the surface makes this activation relatively unimportant. Although activation increases somewhat with power, the problem is not a serious one even at 1 megawatt operation. Short half-life activities are almost negligible in intensity, and long half life activities can be easily removed by small surface purge. In the event the reactor is built at a location where the water sodium content is extremely high, demineralization may be in order. For experiments requiring very low background, demineralized or distilled water may be used, unless the experimentation can be accomplished in a short time following fresh refilling of the pool. Details of water activation are considered further under Shielding, Chapter IV.

5. Critical Mass and Reflector.

Although the critical mass is calculated to be 2.75 kg for a four by four square lattice (12x12x24 in.) with water as a reflector, a somewhat greater amount of fuel (slightly in excess of 3 kg) is required to compensate for losses due to such factors as beam holes and burn-up. There are several other

factors which increase the amount of fuel required in the operating active lattice, including temperature changes, fission product poisoning, and effects due to experimental losses. These are considered in greater detail in Chapter IV, under the subject of reactivity requirements.

The 2.75 Kg of U^{235} is close to the minimum critical mass for this type of reactor. In order to reduce the amount of fuel required as well as to improve the uniformity of the neutron flux distribution within the core, a reflector more effective than water can be placed around the active lattice. For a given power level the neutron density is inversely proportional to volume of the reactor material. Thus, to maximize neutron flux, it is necessary to minimize the fuel volume. This is accomplished by the reflector, which reduces neutron leakage losses. Beryllium-oxide, though expensive, is one of the best, with a lower absorption cross-section than water (.009 barns compared to .660) and a high moderating ratio (183). Because of its high cost, it is necessary to determine the amount to be used in order to gain the greatest returns. A layer on the order of 3 in. accomplishes this. Use of 3 in. of Be-O on the four sides reduces the critical mass to about 2.4 kg, with corresponding lattice dimensions of 9x12x24 in. The Be-O is most conveniently used in the form of cold-pressed bricks placed in aluminum cans of the same shape as the fuel elements, or in flat cans of dimensions equal to the core face, which are easily fitted into aluminum grid which surrounds the active lattice. One such reflector

can be seen in the background of Figure 4.

In Chapter IV, diffusion theory and the two-group theory are used to compute criticality conditions for the 1 megawatt level of operation. It is simply noted here that the approximate results of those calculations indicate a critical mass of 2.2 kg for the water moderated bare reactor, 3.0 kg for the operating reactor, and 1.2 kg with the Be-O reflector. Despite the flux gain from reduced volume, it must be remembered that the smaller the volume the smaller the number and size of specimens that can be irradiated, and the greater relative effect of a specimen being irradiated on the overall flux distribution. This must be accounted for in the form of experimental excess reactivity, normally absorbed by the control rod.

6. Flux and Power Levels.

The fairly wide range of power levels at which this reactor can be operated results in a correspondingly wider range of fluxes available for experimental use. Operated at a low level of one watt, the surface flux will be 10^7 n/cm^2/sec, or 2×10^7 at the center of the core. At 100 kw, a thermal flux density of 10^{12} is available, with an epi-thermal flux of 3×10^{12}. At one megawatt, the average core flux will be 8×10^{12}, increasing to 1.2×10^{13} for an available experimental flux at the surface of the reflector. All of these values will vary somewhat with loading.

It is likely that a reactor used for educational purposes would be run at very low powers, not exceeding a few hundred

watts. At such levels, the accumulation of fission product poisons will be negligible, and the life of fuel elements is almost indefinite. For research work at 100 kw up to 1 megawatt, the life of the fuel elements will decrease accordingly. At higher powers, however, the high flux will reduce the irradiation time necessary for most experiments.

7. Coolant System.

Obviously cooling of the reactor core is essential for proper performance of core materials and maintenance of the integrity of the nuclear fuel. As a coolant, water is quite adequate to provide for the convective removal of heat generated by the low power operation of this reactor. Naturally, the size of the pool, intake temperature of water, etc., are factors determining the degree to which convective heat removal can be used. But for a normal sized pool (say 50,000 gal., or~15x20x22ft) the heat capacity is sufficiently great to permit intermittent operation at the 100 kw level. At this level, with convective cooling, the temperature rise of the water passing through the reactor is about 20°F. Considering all possibilities for variation in temperature (seasonal temperature variations, changing pool water, etc.) the largest variation in temperature to be expected under the most extreme changes in operating conditions is some 50°F.

If steady operation at 100 kw is desired, there will result a rise in pool temperature of 1°F per hour. This must be eliminated either by purging with water of intake temperature (~ 40 gallons/minute is required), or by using a heat exchanger with forced circulation cooling. The latter equipment is necessary if the reactor is designed for 1 megawatt operation. Comments and data on heat transfer requirements are included in Chapter IV.

8. Shielding.

The sources of radiation include the fission process, fission products, radioactivity in the water, and neutron activated members of the structure. Thus three aspects to the problem of shielding need to be considered: 1) the amount of shielding required around the reactor during and after operation, 2) allowable mineral content of the water, and 3) shielding of the lattice framework and the spent fuel elements.

In general the reactor shielding is adequately and simply supplied by water and concrete. Reactor gamma radiation can be reduced to 1/10 tolerance dose rate by 21 ft of water, or 4 ft of water and 8 ft of concrete for 1 megawatt operation; 16;5 ft of water and 7 ft of concrete at pool centerline for 100 kw operation. The water shields the concrete walls from excessive neutron activation (4 ft of water is sufficient). The mineral content of the water can easily be controlled in order to limit water contamination. Shielding from the spent fuel elements and activated reactor structure is provided by placing them in the far end of the pool behind a dam or barrier gate (which retains water when the rest of the pool is drained). For forced circulation systems, the water can be directed away from the surface to allow decay of activated minerals before reaching the surface.

Beam holes can be provided with aluminum canned graphite, concrete, or paraffin plugs. When gaining access to the beam holes, the reactor may be moved to the opposite end of the pool to further decrease the exposure.

Activated minerals in the water include N^{16}, Mg^{27}, Al^{28}, and

Na^{24}. Of these only the N^{16} emits a very high energy gamma (6.7 mev), but its half life is small (7.4 sec). Thus provision is made in the shielding thicknesses, but it is not a problem for residual contamination. All activities except the Na^{24} decay before reaching the surface of the water. The Na^{24} emits gamma of only 2.76 Mev, but with a 14.8 hour half life. The latter is effectively removed by a small surface purge. Apparently such a purge is accomplished by means of a surface run-off, which eliminates the warmer surface layer containing the greatest amount of induced activity.

In the Oak Ridge prototype shielding is accomplished by submerging the reactor to a depth of 16.5 ft below the water surface, allowing 3.5 ft of water between the reactor and the pool floor. Neutron flux is absorbed by the latter distance, but the 16.5 ft is required to reduce gamma flux to safe levels ($\backsim$ 6 mr/hr) for the personnel working hear the pool surface. The concrete walls provide an additional shield, and must be thick enough to afford shielding through the sides of the pool in the event of accidental discharge of the water. Gamma radiation is obviously the deciding factor in determining shield thicknesses, since neutron flux is attenuated with considerable ease. Thus the problem is simply to reduce gamma to a dose rate level below .3 r per week, established by the Radiological Congress (26). (See Chapter IV for additional shielding data).

9. Problems of Pool Use.

The use of water as a triple functioning agent offers many advantages in the form of versatility of use, inherent safety, and low cost. The only two shortcomings immediately evident are relatively minor. They have to do with the corrosion problem, and accidental loss of pool water. The first is a more or less serious problem connected with nearly all reactor designs. The second problem is inherent to the reactor design, and even though its possibility of occurrence is remote, it represents one of the calculated risks which are associated with nuclear reactor operation.

Unless care is taken, the life of the fuel elements of the pool reactor may be determined by corrosion. Rather extensive tests have been performed* to investigate the effects of corrosion, employing pretreatment by anodizing or alodizing, control of pH, and use of inhibitor. The first three were either ineffective, or of limited effectiveness, while the addition of 60 ppm sodium chromate as an inhibitor was found to reduce corrosion to a negligible amount. Moreover, the effect of the Na_2CrO_4 on background activity is also negligible. Such measures may vary depending on the local water supply.

If the reactor core lost its coolant water instantaneously, the reactor would shut down due to loss of moderator. The heat generation due to gamma and beta emission of the fission products drops off to about 6% of the original power, and then decays further by radiation. Assuming instantaneous water loss, one

*Breazeale, (6), pp 45-59.

author's calculation* shows that for an ideal, completely insulated system, the reactor would melt between 16 and 19 minutes after shutdown from 1 megawatt operation. For 100 kw operation the likelihood of melting before reaching equilibrium is almost negligible. However, these results are extremely conservative since the assumptions on which they are based (no heat loss by conduction or convection, and instantaneous loss of water) are difficult to realize physically, short of an earthquake or sabotage. The effect of spreading contaminated water would be difficult to estimate due to the many variables involved (although a possible approach to such an estimate is given by Mills in an article contained in reference (23 pp 417-418). Other aspects of abnormal operation are considered in Chapters III and IV.

10. Reactor Control.

Reactor control must satisfy the dual requirement of facilitating reactor operation to fulfill its function, and of guarding against hazardous operation. The principal danger is development of an excessively high neutron flux, which is primarily a function of the reactivity, and reactor period. With a reactor designed for very low power levels of operation (<10 kw), extended use results in only small changes of reactivity. Such reactors can be loaded so that the condition of prompt critical cannot occur. Thus $k_{ex}=k_{eff} - 1$ is always less than the fraction of delayed neutrons $\sim .006$. Or, in other words, periods are always greater than 4-5 seconds, where the period, T, is defined by the equation $\phi(\vec{r},t) \equiv \phi_o e^{t/T}$, and is the time required for reactor

*Berger, (4), pp 33-44

flux level to change by a factor of e. Obviously the period is not only a factor involved in reactor control, but as such, is fundamental to the stability and safety of the reactor operation. With high power reactors ($\gg$1 megawatt), prompt critical is an easily obtainable condition (i.e. periods can be $<$50 msec). The swimming pool reactor lies between these extremes.

From kinetic theory it is learned that for reactor control purposes it is desirable to maintain the smallest possible excess multiplication factor, k_{ex}. This is readily apparent qualitatively, since it can be shown that when considering only prompt neutrons, the period equation mentioned above becomes $T = \ell/k_{ex}$, where ℓ is thermal neutron mean life-time. Thus with ℓ of the order of magnitude of .001 sec., a change in k of as much as .01 would mean that the neutron flux would increase by a factor of e^{10} in 1 second. It is necessary to prevent such wild excursions of the reactor. It can be shown for example, that a reactivity $\rho = k_{ex}/k_{eff} = +.003$ will result in a period of $\sim$10 sec. Such a value would permit changes in power level over wide levels in reasonable lengths of time, with sufficient margin in avoiding short periods which place severe demands on the reactor safety system.

For 1 megawatt operation it is likely that in loading the reactor and in building it out as burnup progresses, fractional fuel assemblies will be of assistance in maintaining small k_{ex}. In addition it is necessary to consider operation of the reactor by inexperienced personnel. This and other potential accidental increases in k can be estimated to give a value of $k_{ex} \sim 11^{+}$ % for 1 megawatt operation, and a value something less than 5% for 100 kw oper-

tion.* Hence, on the basis of safety considerations which dictate absorbers considerably in excess of maximum k_{ex}, three safety rods (each worth ~6%) are required for the former and two for the latter (~4-5% each).

In considering control, it is in connection with fast periods that concern arises. Safety devices are generally designed to operate at a specified power level (~1.3 to 1.5 nominal full power). However, during start up (with flux ~10^{-9} full power) a very fast period can lead to large overshoots in power. Hence, included in the control design are limited rod withdrawal rate (~1 in./min), and a period scram circuit (which causes the safety rods to be dropped automatically if dangerously short periods develope. Since this reactor has a rather small negative temperature coefficient, automatic servo-control is provided to relieve operators of tedious control tasks, particularly during starting up.

In connection with failure of the 150% power level safeties, it has been experimentally shown that with a slow rise, steady state oscillation at high power level will occur, provided prompt critical is not exceeded. Beyond this level, power fluctuations are more violent, yet still not catastrophic.

The safety control rods contain Boral, a 50-50 mixture of B_4C and 2S Al. The boral is laminated with outer layers of 2S Al, and shaped into an oval cross-section tube loaded with lead (to facilitate gravity drop). The thermal neutron attenuation factor is e^{40}.

* From Abernathy, et al, (1).

Shock absorbers are built into the rods, and a soft iron plate at the top serves as armature of an electromagnet which supports each rod until release by the safety circuit because of accidental or improper operation. The rods fall vertically into special fuel elements which have their central fuel plate removed.

A given power level is maintained by the regulating rod, of similar construction to the safety rods but contributing only .5% to .8% negative reactivity. The limited effectiveness is a requirement of the automatic control feature, since possible failure of this system (resulting in complete rod withdrawal) necessitates limited effectiveness to prevent occurrence of dangerously short periods.

To prevent too rapid a withdrawal of the safety control rods, the speed of withdrawal to the operating level must be limited to approximately 30 minutes for 1 megawatt operation, and 3 minutes for 100 kw operation. For the former this permits a change in k of 1.28×10^{-4}/ sec. with a corresponding shortest period of 92 msec. The latter rate corresponds to a change in k of 2.0×10^{-4}/sec. This is designed to prevent any unsafe overload during start up. Various interlocking devices insure that a negative Δk of 6% to 8% is always available by dropping the rods. Throughout the entire system, interlocks prevent any type of dangerous operation.

The complexities of control, like many phases of reactor design, are such as to make it a large subject in itself. The effects of reactor kinetics inject rather cumbersome non-linear equations into the control problem. Suffice to say that the negative temperature coefficient of this reactor, and the comparative

lack of fission product poisons at low power operation help to simplify the problem somewhat. The details will not be handled further herein, except where they are touched upon in Chapter IV.

Control Circuits. The electrical control circuit for the pool reactor contains three differential ionization chambers whose current outputs are a measure only of neutron flux. These consist of 2 chambers connected in opposition, one of which is boron coated; both respond to gamma, only the boron chamber to neutrons. One differential chamber supplies a galvanometer whose reading is proportional to chamber current to a range of 2×10^4. Another feeds a micromicro ammeter which, with the aid of shunts, extends the power range to 2×10^5. This second chamber controls a Brown Recorder which drives the servo-amplifier. A third chamber feeds a logarithmic amplifier and controls another Brown Recorder (the Log N indicator) giving a continuous record of power level over a range of 10^6. A pile period indication is obtained by differentiating the amplifier output.

For startup, a small neutron source ($\sim$3 counts/sec) is used, with a U^{235} fission chamber, amplifier, scalar, and register. The starting circuit covers the range from 10^{-4} to 1 watt, and registers the neutron flux from the starting source to a level above critical. The fission chamber is mechanically withdrawn to prevent activation at high reactor powers.

Because the system must "fail safe" (safety rods must fall if power or circuit trouble develops) vacuum tubes are used throughout the safety system instead of gas tubes or relays. This system includes 2 boron-coated ionization chambers which supply a current

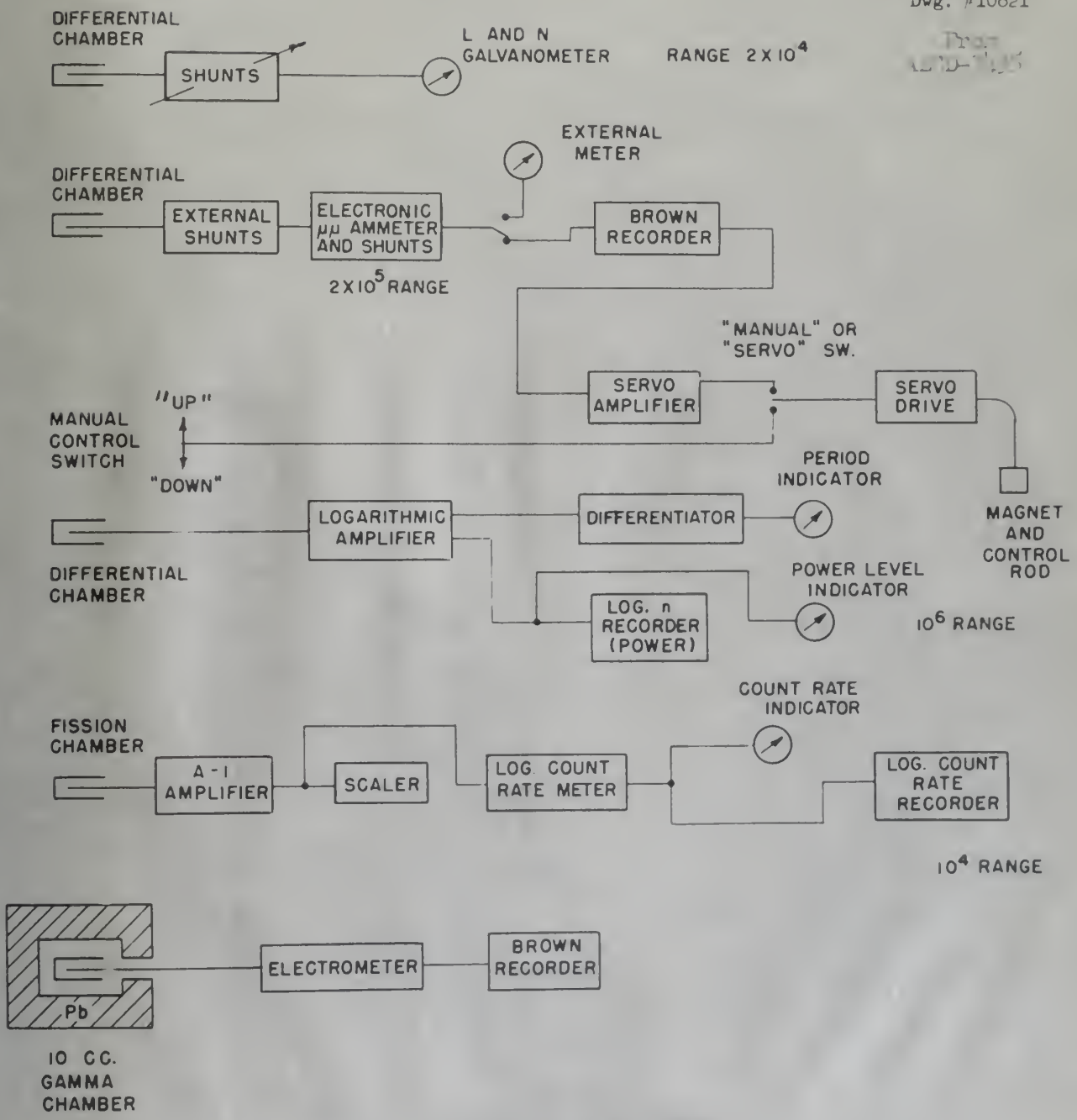
proportional to the neutron plus gamma level. This current flows through the high resistance of a pre-amplifier (single stage cathode follower) and then to a DC amplifier which supplies the signal for operating the safety circuits. (See Figures 5 and 6 for block diagrams of control and safety circuits.)

All of the control and monitoring electronic equipment can be contained in five relay racks, three forming the control panel, and two containing power and amplifiers for the servo-control mechanisms. Reliable package controls, experimentally proven through use with the prototype model, are commercially available. Some improvements toward simplification have already been recommended (1), however, and indicate that use of a pool reactor at the 1 megawatt power level would justify improved instrumentation. Considerable material is available in the unclassified literature concerning details of control systems for low power research reactors.*

11. Construction Features and Auxiliary Equipment.

The simplicity of the basic swimming pool reactor design is such that it lends itself easily to modifications in the way of shape, size, location of experimental equipment, and so on. For example, one of the best designs employs a hillside location which offers the following advantages: 1) lower shielding costs, 2) less excavation required for pool, 3) easier drainage, and 4) easier access to beam holes and experimental equipment. Wherever the location, there are no significant problems to contend with in build-

*Abernathy (1), Breazeale (6), Cole (8), and Harper (18).

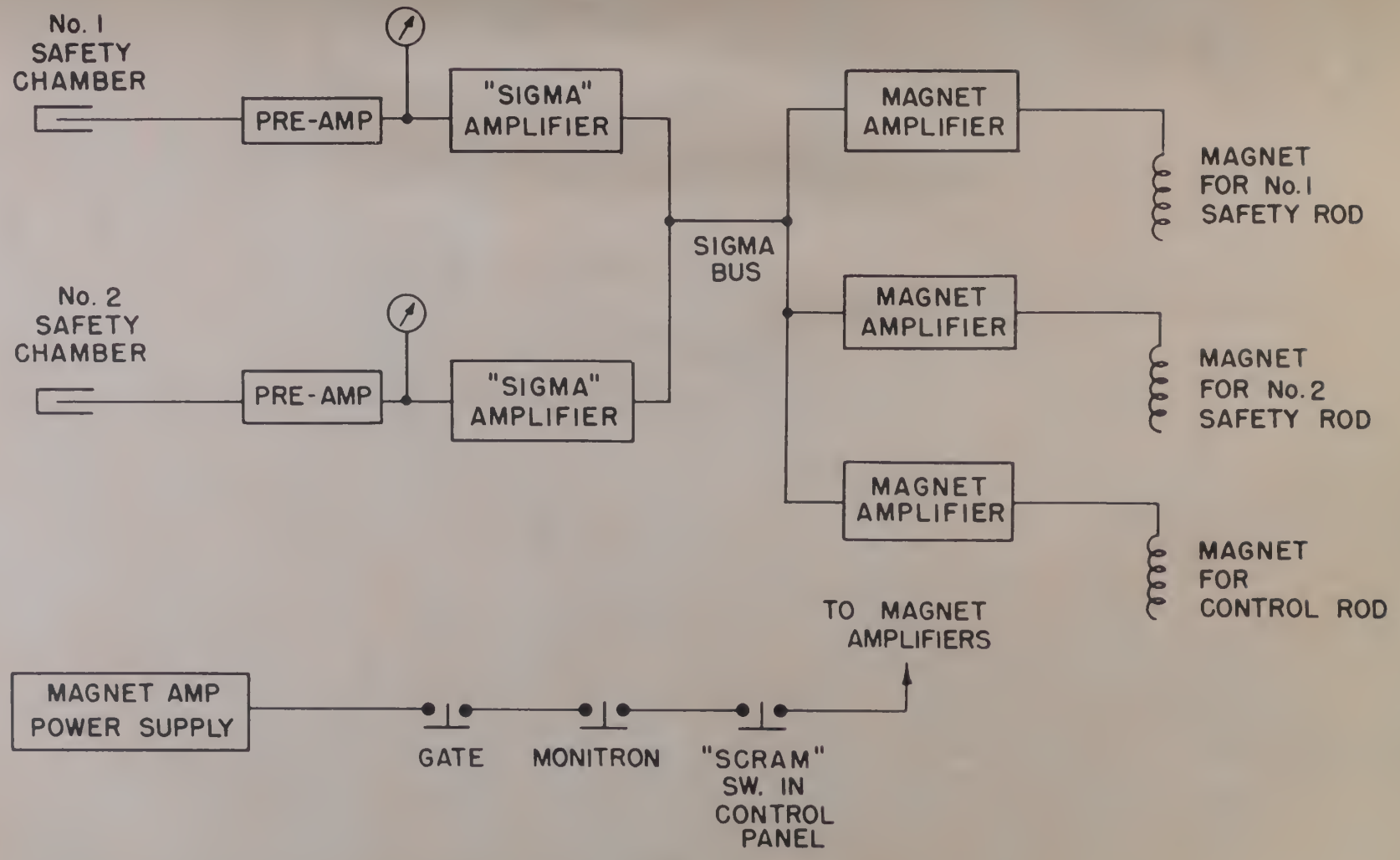


CONTROL CIRCUITS

Fig. V

Unclassified
Dwg. # 10822

From
AECD-3435



SAFETY CIRCUITS

ing the structure to house the reactor. It can be as large or as small as needs or available funds dictate. The sole requirement of substantial construction for both shielding purposes, and to withstand heavy equipment loads is the only major construction cost. The building can be of steel frame construction with sufficient head room for a 30 to 50 ft. free lift. This will permit the use of a 1½ to 2½ ton crane for placing fuel elements, positioning pool gate, instruments and samples, and performing maintenance. Cement flooring is necessary to bear heavy loading. Except for one set of 8x10 rollaway doors at the ground floor level, all other doors, stairs, and windows may be standard.

A concrete pool (steel reinforced) can be of rectangular, pentagonal, or hexagonal shape, depending on the type of research to be accomplished, or the desired positioning of beam holes. The prototype has a 40x20x20 ft. pool, but a pool as small as 14x18x22 ft. could suffice. The latter is the pool size of the design evaluated in the cost section of this thesis, including a bay 28x28x30 ft. high, and 2500 sq. ft. of laboratory and office space. Pool slope for drainage, and a carry-off drain to permit skimming the surface and to provide a 4-5 gal/min overflow (to eliminate ripples for better visibility) are essential features.

The bridge can be constructed of a pair of steel I beams with transverse bracing, and covered with a wooden platform. One side of the bridge suspension should be open, permitting operation of controls and placement and removal of objects in and around the reactor.

Beam holes are relatively inexpensive luxuries, and one to

three are recommended. These can be constructed of aluminum ½ in. piping of 6 to 10 in. inside diameter, closed end against the face of the reactor, with water tight flanges fastening through the pool wall, and using gate valves on the inner face of the pool wall.

A dam or gate near one end of the pool is necessary to retain water to shield the reactor frame and the fuel elements when not in use. This is necessary whenever the main portion of the pool has been drained for maintenance or placement of samples.

If it is desired to use the reactor often or continually at powers of 100 kw and above, forced cooling with the attendant pumps and heat exchanger are necessary. The previously suggested purging for steady 100 kw operation, although effective, is a problem because of the expense of adding sodium di-chromate to the water.

These are the basic design requirements. Over and above these, improvements will depend on the scope of the instructional and research programs undertaken.

CHAPTER III

CHARACTERISTICS, USES, AND PROBLEMS OF REACTOR USE

1. Versatility of Use.

The basic simplicity of design of the swimming pool reactor is responsible for many of its advantages with regard to its versatility. With any research reactor there are three major problems to contend with: 1) accessibility of the reactor structure, 2) availability of beam holes, and 3) adequate space for experimental equipment. The swimming pool has features which overcome many of the difficulties related to these three considerations.

Accessibility of the reactor structure. A feature that is unique to the swimming pool is the ability of personnel to work on the reactor structure a mere few hours after the pool has been drained, and the fuel elements and control rods are removed. The convenience of pumping the pool dry while reactor frame and fuel elements are still shielded in the water behind the barrier gate or dam facilitates repairs and adjustments, and the placing of instruments and samples to be irradiated. The fact that the fuel elements are removable and storable in water behind a barrier makes maintenance of the reactor structure itself an easy matter. This flexibility is of unlimited value for educational purposes, for with a minimum of difficulty it enables different groups of students to assemble and dissassemble the reactor on successive days or even in the same day. This provides complete familiarization with operation of all controls, and makes possible repeated experiments on

criticality and effects of different loading geometries.

The beam hole. One of the greatest problems with other reactor designs is the provision of beam holes in the active lattice, into which the experimental samples and instruments are placed. In becoming acquainted with conventional reactor design, it is readily apparent that to make provision for beam holes is an awkward problem. It is the inclusion of such features that rapidly increases the cost of reactor construction. Moreover, the space available in the holes is limited, they must be carefully plugged, and there is no direct access to the beam hole because of the intense activity.

The swimming pool provides what amounts to an infinite beam hole, since equipment, instruments, and samples can be lowered into the pool, and positioned against the reactor face. Experience with the Oak Ridge model has revealed that the underwater use of instruments and equipment is both simpler and safer. To a certain extent the principle is universally applied since it is normal procedure with most reactors to have a water-filled reactor canal for safe-handling and storage of spent fuel elements, and most research and handling operations where it is necessary to see what is being done are most easily handled under water. With the pool reactor, the core and control mechanism can be visually observed at all times during operation.

When a collimated beam is required, a conventional beam hole can be provided in the pool reactor with only a small increase necessary in the critical mass. This, again, is facilitated by the extreme flexibility of the components of the active-lattice.

Instead of the usual problems connected with construction of a beam hole into the reactor core itself, the pool reactor is simply moved into position against an empty closed-end aluminum tube. The closed end fits flush against the reactor (or reflector) face, and the opposite end is bolted against the inner pool wall with a water proof gasket to prevent leakage of pool water. An alternative to the horizontal arrangement is a vertical arrangement with the end extending above water. There is adequate space for several beam positions. The comparative ease of installation should be apparent.

Experimental space. With most research reactors, the shielding material and control equipment is of such bulk, or must be so positioned that much of the valuable experimental space immediately adjacent to the reactor is lost. From the experience gained by researchers working with some of the early reactors, basic design criteria have been tabulated for guidance.* The pertinent point, however, is that the swimming pool again offers several advantages. The total pool dimensions are available for placement of instruments and samples, against the faces of the active lattice, at the ends of collimated beams, in the water, or in the air (with the reactor behind the barrier). In addition, the bottom surface of the pool can be constructed of removable blocks, providing an adjustable floor level, and additional flexibility in placing samples and instruments. For investigation of neutron distributions, foils are placed in the small gaps which exist between the parallel rows of fuel elements. If

*Goodman (18), and Glasstone (16).

irradiation or instrument readings are desired within the core structure, it is possible to place samples and specially designed instruments in dummy fuel cans, rearranging the loading geometry accordingly.

Power range. The range of powers and corresponding fluxes available from this reactor offer a moderate choice to anyone interested in this reactor type. The power level selected will depend largely on its intended use, with the very low range having its primary value as an educational facility, the 100 kw and above providing a research flux density (10^{12}) which compares favorably with that of the large graphite reactors available today, and the maximum of 1 megawatt extending the potential field of specialized research somewhat farther.

2. Safety of Design.

Perhaps the safest ideal reactor design would be a thermal, enriched, homogeneous, solid core type, with a large transient negative temperature coefficient. Naturally, a lower power helps to increase safety. To have a low power with a high flux, a low critical mass is necessary. This is evident from the relation $\phi = K \cdot P/W_f$, or, flux is directly proportional to power, inversely proportional to the weight of fissionable material.

The swimming pool design is close to the ideal, then, as an intrinsically safe reactor design. Its only evident departures from the ideal are its rather small negative temperature coefficient, and the safety problems associated with a pool of slightly contaminated water. The latter shortcoming is far out-

weighed by the many advantages of the pool design. And, in general, any of its theoretical shortcomings are overcome by compensating measures which make it basically a very safe reactor.

Given the basic design, there are two very general aspects to safety considerations. One involves the safety of reactor personnel while the facility is operating normally, and the other is concerned with safety features which control dangerous conditions and potential accidents. Both aspects involve overall reactor design, and both presuppose a complete program of health physics protection.

Normal operation. As already mentioned, water and concrete provide adequate shielding for personnel and equipment during normal operation. (It is interesting to note that since the neutron flux is easily attenuated through elastic scattering with the hydrogen of the water, the thickness of the water shield is actually determined by gamma tolerances. Thus, variations in reactor design could greatly reduce the thickness of the water shield by use of only a few inches of iron or lead; although the desirability of such a modification for any but a special purpose design is doubtful.) With a good health physics program, and with adequate instrumentation, there is no problem with accessibility of the pool area a short time after shut down, or in insuring that personnel working normal hours around the reactor will stay well below the .3 r per week tolerance dose.

Abnormal operation. Although the design negative temper-

ature coefficient of -.0075%/°F exercises a small stabilizing control on the order of ½% in reactivity, the true safety control over reactivity comes from the experimentally observed fact that arise in power level to more than a few hundred kw will cause boiling. The steam then displaces sufficient moderator to prevent runaway. This conclusion is based on calculations of kinetic behavior after stepwise increases in k_{eff} of 2% (from steady state powers of 1 kw) above prompt pritical. The resulting fuel plate temperature (maximum) was 390°F, the critical condition persisted until 10% of the moderator was expelled by steam, and the reactor became subcritical in 0.127 seconds.* Although not conclusive, the calculations are indicative of a fairly reliable design safety factor. Further calculations have indicated that boiling will make the reactor subcritical before the fuel plates are damaged, and the reactor will then oscillate at an average power level of ~200 kw. In that case, personnel standing near the pool would not receive an overdose. Augmenting this are two facts: 1) the fuel plate design is such that thermal stress is not a factor (the stability under calculated conditions of water loss, Section 9, Chapter II, and the heat transfer data of Chapter IV testify to this); and 2) the "fail Safe" control of the reactor provides a wide margin of safety in the prevention of runaway. The safety controls system is arranged to shut down the reactor when neutron flux levels rise to a point indicating an overload

*From Beck (3).

of approximately 30% (at 100 kw) to 50% (at 1 megawatt).

In the event of unforseeable conditions under which the pool water may become more heavily contaminated than normal, the water will have to be closely monitored before permitting re-entry to lake, river, or normal outlet. This includes knowledge of the radio-chemical composition of the released water, and measurement of minimum dilution assurable in the receiving outlet. Normal operation results in such low levels of activity that the direct discharge into normal outlets is quite feasible.

Other contingencies, including radiation from an unshielded experimental reactor beam hole or from handling of an irradiated object with insufficient shielding, fall in the category of monitoring safeguards and implementation of a rigid health physics program. (Morgan (25) provides an excellent coverage of the health physics considerations, including guidance in laboratory design.)

3. Cost.

The cost of a research reactor such a s this can be divided roughly into three categories: cost of reactor and controls, housing structure for the reactor, and cost of uranium. On the basis of the Oak Ridge prototype, the reactor and controls cost approximately $60,000. Necessary auxiliary equipment (reflector, spares, and instruments) an additional $35,000.

The size of the reactor design and related equipment warrants the use of a separate building to house it. Naturally, the cost will vary somewhat depending on the design, materials, and local conditions. One estimate*, based on the cost of a

*Breazeale (5)

14x18x22 ft pool, a 28x28x30 ft bay containing the pool, and 2500 sq. ft. of laboratory space is $125,000. The Oak Ridge prototype has a building with a bay 70x30x38 ft. high, holding a 40x20x20 ft. pool, with an additional 3000 sq. ft. of office and shop space, costing $137,000 in 1950. A design modification of this structure, using original prices for estimates, and increasing the facilities slightly for a 1 megawatt reactor, was estimated at $167,000. The latter model would have an increased cost of $15,000 for pumps and heat exchanger, plus a few thousand dollar increase to cover spares and heavier equipment. The total cost (building, reactor, and controls) being approximately $270,000. As a rule of thumb, then, reactor and controls without building $100,000; with building, an additional $125,000 or more depending on the scope of the program, and current construction costs.

It will be noted that the Be-O reflector is one of the major items of cost (~$17,000). In view of previous comments, however, this cost must be weighed against AEC annual inventory charges on the uranium which is saved through the use of this reflector.

Facilities for experimental projects must be added to the above, along with the expense of salaried personnel for operating the reactor.

Uranium costs have been subject to wide fluctuations as a result of changing policies within the AEC. The problem is one which has many political as well as technical ramifications. According to the McMahon Act, the AEC is directed "to distri-

bute sufficient fissionable material to permit the conduct of widespread independent research and development activity, to the maximum extent practicable." The degree to which AEC will be able to support academic institutions in this program is still a pending matter, awaiting determination of an over-all government policy.

It is difficult to compare costs of different types of research reactors, in view of the fact that each one is constructed to suit specific needs. One school plans to build a water boiler type for $50,000. The North Carolina reactor cost ~$100,000. Both figures are for reactor and controls only. The North American water boiler is estimated at $75,000 to $100,000 depending on the type of structure in which the reactor is housed. The Norwegian 100 kw heavy water reactor cost approximately $180,000 for reactor and controls, plus an additional $2,000,000 for heavy water; indicating the wide variance in cost for different designs and materials of construction. Certainly, for the scope of research possible with the swimming pool reactor, it is among the least expensive of the research reactors. (Note: the MTR, Materials Testing Reactor, at Arco, Idaho, is also a thermal, heterogeneous, light water research reactor producing both fast and slow neutrons, with a flux 4×10^{12}. It is being used for classified research in shielding materials exposed to intense radiations. Cost: ~$1,000,000.)

The following table is included as a guide to provide some idea of the types of auxiliary equipment required, and order of magnitude costs.

TABLE III-BREAKDOWN OF REACTOR COSTS*

CONTROL AND INSTRUMENTATION COSTS

Instruments and Accessories

1 Micro-micro-ammeter, Leeds & Northrup #9836A	$1,150.00
1 Shunt & mounting for micro-micro ammeter	300.00
1 Log n recorder, single point, Brown	650.00
1 Counting rate recorder, single point, Brown	650.00
1 Micro-micro-ammeter recorder, single point, Brown	650.00
1 Log n Amplifier, type Q-915	725.00
1 Log Counting Rate Meter, type Q-751	425.00
1 1024 Scaler, Atomic Instrument Co.	700.00
1 Amplifier, A-1, type Q-541	400.00
1 Fission chamber & preamp for underwater op,type Q1059	500.00
2 Ionization chamber, parallel circ plate, type Q-975	1,650.00
2 Ionization chamber, compensated, type Q-1045	1,700.00
2 High Voltage Supply, type Q-995	900.00
Subtotal	$10,400.00

Control System

3 Safety Rod and shock absorber assy	2,250.00
1 Regulating rod	150.00
3 Magnet assy	1,500.00
3 Magnet guide	150.00
1 Safety rod drive motor	50.00
1 Geering, Pulleys, Music Wire, etc.	150.00
1 Limit switch assy, and Selsyn for safety rods	125.00
1 Servo Amplifier	600.00
1 Servo Motor, gear train, tachometer, Selsyn, limit switch, etc.	625.00
Subtotal	$5,600.00

Miscellany

4 Voltage reg transformer, 250 VA	300.00
Relay cabinet with relays and console, internally wired	3,000.00
Amplifier racks and wiring, terminal boards, cable, conduit, etc.	1,000.00
Gas flow equipment for underwater instrument heads	500.00
Assembly and testing	2,500.00
Subtotal	$7,300.00

Spares	7,375.00
Reserve for contingencies	4,000.00
H. P. Instruments	7,500.00
Subtotal	$18,875.00

(Cont'd):

*From Abernathy (et al), AECD 3557 , (1).

Building Costs

Excavation of 1047 cubic yards @ $9.50 per		
based on composition of 0.9 dirt and 0.1 rock		
dirt removal @ $5.00 per yard		
rock removal @ $50.00 per yard		9,950.00
Pool concrete		
bottom 23 yds. @ $25.00		
sides 139 yds. @ $85.00		12,390.00
Building		
total volume 130,280 yds @ $1.00		130,280.00
Crane - 38 feet 5 ton capacity		7,000.00
Concrete blocks for shielding (82.5 yds in place required)		6,100.00
Beam hole plugs - 6 required @ $100		600.00
Inner beam hole plugs - 3 required @ $100		300.00
	Subtotal	$166,620.00

Reactor

20 Fuel elements (including the partial elements)@120		2,400.00
Reactor bridge		2,160.00
Lower fuel grid and the superstructure		2,040.00
*Beryllium-oxide reflector elements if desired 14 @ $747		10,458.00
Assembly of above		1,000.00
*Will reduce number of fuel elements required	Subtotal	$18,058.00

Equipment for Forced Cooling

Heat Exchanger		3,400.00
Lower primary pump (1000 gpm, 50 ft head)		2,000.00
Secondary pump (1000 gpm, 35 ft head)		500.00
Transition piece and bellows		1,500.00
Valves 6 @ $250		1,500.00
Piping and fittings		2,000.00
Installation of the above equipment		4,000.00
	Subtotal	$14,900.00
Engineering Fee - 12%		$29,010.00
	GRAND TOTAL	$270,763.00

4. Uses of the Research Reactor.

The uses of a research reactor are extremely numerous, providing an intriguing list of research projects for several fields, with emphasis on the physical, chemical and biological. As a prolific source of neutrons, the reactor, if operation is properly supervised and scheduled, can be used to carry on several phases of research simultaneously. Where the experimental requirements call for thermal neutrons, a thermal column can be used. Similarly, other beam holes can be used in conjunction with specially adapted velocity selectors to provide beams of narrow energy spread. Although complete exploration of some research projects is not possible with low power operation, partial exploration will be. The range of neutron flux provided by the pool reactor is considerably greater than that of many research facilities, permitting a correspondingly wider scope in the research program. The reactor can also serve as a source of fission products through activation of uranium salts, as well as a source of activated isotopes which serve as powerful tools in achieving other research goals. Many such research problems are both vital and timely with regard to AEC interests and national defense. For example, within the field of chemistry, investigation of the fission product distributions resulting from differently initiated fission processes might provide greater understanding of the fission process itself. The study of polymerization effects of radiation, already lending understanding to certain problems with the Geiger Counter, and electron microscopes, has many more far reaching aspects to investigate.

Atomic recoil phenomena presents a fertile research field, leading into any one of several special fields, of which activation analysis is one. The production of artificial isotopes by the (n,γ) reaction has opened up a broad and vital field in radio-chemical analysis.

In biological research a new approach to radiation sickness and its treatment has been made possible through the study of the effects of radiation on biochemicals. A great deal more investigation needs to be done on the physiological radiation effects in animals. Much of the latter (injury threshold, lethal dosage, effects on body organs, etc.) is significant with relation to atomic warfare and civil defense.

In physics the possible investigations into nuclear, phenomena are too numerous to mention in any detail. In addition to studies of nuclear phenomena, much of the development of reactor research techniques and instrumentation lies within the field of physics. This point is relative, however, since specialized research will stress the ingenuity of all the individuals involved, whatever their special field.

Not to be overlooked are the possible uses of the research reactor for medical and industrial purposes. These are largely a factor of proximity of using agencies to the reactor site, and because of this may not be a consideration of an institution such as the United States Naval Post-Graduate School. The possibility of providing either tracer isotopes for medical use, (or even a portion of the lab space for X-ray or neutron beam

therapy), or irradiation of industrial parts for materials testing, may be considered as a means for defraying the cost of a reactor. Such a plan might even make possible the installation of larger facilities so that the other portions of the research program will not suffer from these additional projects.

In spite of the significance of the research aspect, it is likely that the reactor has an application for a graduate engineering institution which is as important or more so than research. This is the use of the reactor for instructional purposes in the study of reactor theory, design, and operation. The same virtues in the way of versatility of use already outlined, make the pool reactor perhaps the most effective one available for training purposes. Other reactor types, because of their bulk and inaccessibility are rather rigid, once constructed. They are then available for research and limited instruction in reactor operation, and study of neutron flux distributions. However, students (other than those few who may participate in its original construction) are not permitted to acquire first hand familiarity with many of the reactor parameters. The flexibility of the pool reactor would permit adaptation to almost any phase of study in reactor theory, including a modified exponential experiment. Moreover, the two programs, research and instruction in nuclear engineering, could easily be integrated by careful scheduling of the related activities.

Some of the specific uses of this type reactor have been outlined on the following pages. The list is suggestive in length, rather than comprehensive.

TABLE IV

SUGGESTIVE LIST OF REACTOR USES

I. Instructional in Reactor Operation

A. Study of reactor theory and design.

1. Performance of Criticality experiments
2. Investigation of neutron distributions
3. Effects of different loading geometries
4. Space and velocity distributions of radiation
5. Effects of temperature on reactivity
6. Investigation of transient phenomena
7. Investigation of inhomogeneity effects
8. Exponential experiment

B. Study of Reactor Controls and Instrumentation

1. Assembly of reactor core
2. Operation of control system
3. Use of experimental equipment - beams holes, velocity selectors, crystal spectrometer, etc.
4. Use, calibration, and installation of health physics instruments.

II - Research Uses

A. Physical

1. Investigation of microscopic nuclear phenomena

a. Threshold energies of nuclear reactions

b. Nuclear cross-sections

c. Multiple product yield ratios

d. Properties of radiation; polarization, neutron decay

(Cont'd)

e. Resonance Absorption characteristics

f. Delayed neutron study

g. Heat transfer and power studies

h. Neutron diffraction

2. Investigation of Macroscopic Nuclear phenomena

a. Attenuation and transport of radiation of various types and energy in different materials

b. Radiation reflectivity

c. Radiation induced luminescence and flourescence

d. Effects of radiation on thermal and electrical conductivity

e. Determination of crystal structure by diffraction

3. Development of techniques for separation of neutron energies

4. Radiation effects on materials

a. Shielding studies

b. Radiation damage to materials

5. Tracer Studies

a. Production of isotopes

b. Industrial applications.

B. Chemical

1. Product distribution differences for differently initiated fissions

2. Effects of irradiation of chemicals

3. Polymerization effects of radiation

4. Atomic recoil phenomena

(Cont'd)

5. Analysis by radioactivation

6. Identity and characteristics of fission products

C. Biological

1. Radiation effects in animals

a. Injury threshold

b. Lethal dosage

c. Effects on body organs

d. Genetic effects

2. Radiation effects on bio-chemicals

3. Tracer studies

4. Medical applications

a. Use of isotopes in clinical studies

b. Boron localization in metastic tumors

c. Possible X-ray source

5. Problems of Reactor Operation.

In the foregoing material the problems associated with the reactor design, construction, use, safety, and initial costs, have been discussed in detail. There are two more significant problems associated with the operation of a reactor. They are: 1) availability of skilled technical and supervisory personnel, and 2) additional costs of continued operation.

Naturally it is imperative that the responsibility for a reactor program be entrusted only to persons who have had training and experience in the actual operation and behavior of nuclear reactors. The operation of a reactor will require the full time efforts of several specialists, the part time

efforts of a large number of the teaching staff of the physics, mechanical, electrical, and chemical engineering departments, and a small staff of full-time technicians. The requirement for the specialists is evident when realizing that only a few of the major activities requiring personnel are:

1. Operation of the reactor and related facilities
2. Instruction in reactor operation and related activities
3. Maintenance of equipment - reactor, pumping and heat exchanger, electronic controls, etc.
4. Health physics activities - safety program, instrument maintenance
5. Laboratory assistants for research and training programs
6. Special services - accountability for fissionable materials, classified documents, security

The above listing is not complete, nor is much information available on the numbers of personnel required and costs involved in continued operation, due to the fact that low cost research reactors run on the relatively modest scale of an academic institution, have not long been in operation. The matters are mentioned here since they are inherent to reactor operation, and any feasibility study will have to consider the problem as one of major importance.

CHAPTER IV

THEORETICAL CONSIDERATIONS

1. Criticality Calculations.

General. Employing the reactor theory of several of the basic texts*, together with the data and approximating techniques contained in AECD-3557 (1), the critical size of the active lattice for 1 megawatt power level and the corresponding fuel requirement can be calculated. Before undertaking zero power criticality experiments, it is desirable to know the lower limit of the critical mass required in a clean (no beam holes, control rods, or experimental equipment) unpoisoned core, with only water as a reflector. This calculation can then be modified to give the similar results for an operating core.

Since exact calculations of this sort are virtually impossible, there are permissable simplifications which may be made without involving errors even as large as those introduced by the uncertainty in the fundamental moderator constants. The most important simplification is that of substituting an equivalent homogeneous system (of U^{235}, Al, H_2O) for the heterogeneous system.** This resolves the problem into that of solving for the macroscopic variations represented by "smooth" functions of homogeneous mixtures (which give close approximations to the desired results) while neglecting the local microscopic variations over distances small compared to the dimensions of a lattice cell. Solutions involving the latter will

*Glasstone (16), Soodak (17), and Goodman (32)
** See Section 3, p 60, for justification

be mentioned at the end of this section.

A further simplification is introduced by assuming an infinite water reflector (effectively so because of the thickness of surrounding water compared to the neutron diffusion length). The assumption of cylindrical geometry for the core, the results of which are converted to the equivalent square rectangular dimensions, is introduced as another labor saving device.

Finally, the rather complex mathematics involved in a finite nonspherical geometry two-group diffusion problem is greatly reduced by closely evaluating the spatial flux in a given direction, and assuming a simple asymptotic flux in the dimensions perpendicular to the given direction. This amounts to determining the axial flux assuming infinite reflector in the axial direction, while unreflected in the radial direction; the reverse condition is assumed for the radial flux. The failure to account for certain reflector regions introduces only a small error, since those regions are several reflector diffusion lengths from the core.

Notation and constants. In the calculations to follow, the following notation and conventions will be used. The moderator will be understood to be the aluminum-water mixture of the unit homogeneous cell. Terms without subscript refer to thermal values; "1" refers to fastgroup neutrons, and "2" to the thermal group; "R" indicates the region of pure water reflector, subscript "M" the moderator region, and "C" the core; in lower case letters "c" refers to non-fission capture, "f" to fission, "a"

total absorption, "s" scattering, and "tr" transport scattering.

The basic constants employed are listed in Table V, page 5D. These constants are used in the preliminary moderator calculations of the next section, namely the calculation of macroscopic cross-sections, Σ TR(M), and Σa(M), diffusion coefficients D_{2M} and D_{1M}, and diffusion area L^2_M, under the assumption that the scattering and transport properties of the core are those of the Al-H_2O mixture alone (i.e. $D_{1C}=D_{1M}$, $D_{2C}=D_{2M}$, $\tau_C=\tau_M$).

TABLE V - NUCLEAR CONSTANTS* AND PARAMETER VALUES.

Cross-Sections for 0.025 ev thermal neutrons: *

Miscoscopic:		Macroscopic:	
$\sigma_{tr}(H_2O)$	= 73.8 barns	$\Sigma_{tr}(H_2O)$	= 2.47 cm^{-1}
σ_{tr} (Al)	= 1.4	Σ_{tr}(Al)	= 0.084
σ_a (H_2O)	= 0.64	$\Sigma_a(H_2O)$	= 0.0214
σ_a (Al)	= 0.22	Σ_a(Al)	= 0.01325
σ_f (U^{235})	= 549	$\Sigma_f(U^{235})$	= 0.0649
σ_c (U^{235})	= 101	$\Sigma_c(U^{235})$	= 0.0119

Reflector Constants (water 20°C)*:

L_R^2 = 6.30 cm^2 (Thermal Diffusion Area)

D_{2R} = 0.135 cm (Thermal Diffusion Coefficient)

D_{1R} = 0.9 cm (Fast Diffusion Coefficient)

τ_R = 33 cm^2 (Fermi Age of Thermal Neutrons from Fission sources)

Calculated Moderator Constants:

L_M^2 = 8.92 cm^2 $\quad\quad$ $\Sigma_{tr(M)}$ = 1.918 cm^{-1}

D_{2M} = 0.174 cm $\quad\quad$ $\Sigma_{a(M)}$ = 0.0195 cm^{-1}

D_{1M} = 1.1 cm

τ_M = 45.5 cm^2

For the Unpoisoned Core:	For the Operating Core:
L_c^2 = 1.81 cm^2	L_c^2 = 1.802
k_∞ = 1.685	k_∞ = 1.513
$D_{1c} = D_{1M}$	
$D_{2c} = D_{2M}$	
$\tau_c = \tau_M$	

* From Adair, Rev Mod Phys., Vol. 22, pp 249-289, 1950; and TID - 235, Nuclear Data for Low Power Research Reactors, Nov 1950; and references (1) and (16)

Preliminary calculations. Employing the exact fuel element and grid measurements, a unit cell is found to have dimensions of 3.035 in. x 3.189 in. x 25 in., including half of the water layer between elements, and $\frac{1}{2}$ in. water at each end. The resulting cell volume is 3965 cm^3, with aluminum to water volume ratio of 0.3. The fuel economy design (based on optimization of the fuel to moderator ratio) calls for 183 gm of U^{235} per cell. The concentration of U^{235} is then calculated to be 46.15 gms/liter. Fuel enrichment (> 90% U^{235}) is sufficient to warrant the neglect of U^{238} in the calculation.

Letting α = Volume Al/ Volume H_2O in the core = 0.3, then the volume fraction of aluminum and of water in the core are $\alpha/1+\alpha$ = .231, and $1/1+\alpha$ = .769 respectively. Using these values and the cross-sections in Table V the following moderator constants are calculated:

$$\Sigma_{tr(M)} = \frac{1}{1+\alpha}\Sigma_{tr}^{H_2O} + \frac{\alpha}{1+\alpha}\Sigma_{tr}^{al} = 1.918\ cm^{-1}$$

$$\Sigma_{a(M)} = \frac{1}{1+\alpha}\Sigma_{a}^{H_2O} + \frac{\alpha}{1+\alpha}\Sigma_{a}^{al} = 0.0195\ cm^{-1}$$

$$D_{2M} = \frac{1}{3\Sigma_{tr(M)}} = \frac{1+\alpha}{3\left[\Sigma_{tr}^{H_2O} + \alpha\Sigma_{tr}^{al}\right]} = 0.174\ cm$$

$$L_M^2 = D_{2M}/\Sigma_{aM} = 8.92\ cm^2$$

$$D_{1M} = \frac{1+\alpha}{3\left[\Sigma_{1tr}^{H_2O} + \alpha\Sigma_{1tr}^{al}\right]} \approx \frac{1+\alpha}{1/D_{1R} + 3\alpha\Sigma_{tr}^{al}} = 1.1\ cm$$

The last relation assumes that the average scattering and transport crossSections for aluminum in thermal and epithermal regions are approximately the same. τ_M (taken from the published literature) and the above values are entered in Table V for ease of reference.

For calculations of the multiplication factor, k, the following quantity is useful:

$$z = \frac{\Sigma_u}{\Sigma_M + \Sigma_p}$$ where $\Sigma_u = \Sigma_f^{U^{235}} + \Sigma_c^{U^{235}} = .0768\ cm^{-1}$ is the initial macroscopic total absorption cross-section for U^{235} in the core, and Σ_p = Σa (due to poisons) = o for clean core, or $z = 3.941$ for initial clean core.

For an infinite array of fuel elements, $k_\infty = \eta f p \epsilon$ where

f = thermal utilization = $\frac{\Sigma_u}{\Sigma_u + \Sigma_M + \Sigma_p} = \frac{z}{1+z}$

η = average # of fast neutrons emitted/thermal neutron capture in fuel

$= \nu\, \Sigma_f / \Sigma_u = 2.5 \times 549/650 = 2.11$

p = resonance escape probability = 1 } for enriched, thermal homogeneous core

ϵ = fast fission factor = 1 }

Thus:

$K_\infty = \frac{2.11\, z}{1+z} = 1.685$ for initial clean core, and

$L_c^2 = D_c / \Sigma_c \cong \frac{D_M}{\Sigma_u + \Sigma_M} = \frac{L_M^2 \cdot \Sigma_M}{\Sigma_u + \Sigma_M} = \frac{L_M^2}{1+z} = 1.81\ cm^2$

for the thermal diffusion area of the clean core.

Two Group Diffusion Method. Continuing with the analysis as outlined in Glasstone (Chapter VIII), the solution of the fundamental flux equations are obtained for the two group model. For the core these equations are:

$$\left.\begin{aligned} D_{1c} \nabla^2 \phi_{1c} - \Sigma_{1c} \phi_{1c} + k \Sigma_{2c} \phi_{2c} &= 0 \\ D_{2c} \nabla^2 \phi_{2c} - \Sigma_{2c} \phi_{2c} + \Sigma_{1c} \phi_{1c} &= 0 \end{aligned}\right\} \qquad (1)$$

Or, since they are more conveniently handles in operator form:

$$\left.\begin{aligned}(D_{1c}\nabla^2-\Sigma_{1c})\phi_{1c}+k\Sigma_{2c}\phi_{2c}&=0\\ \Sigma_{1c}\phi_{1c}+(D_{2c}\nabla^2-\Sigma_{2c})\phi_{2c}&=0\end{aligned}\right\} \qquad (2)$$

If non-trivial solutions are to exist for ϕ_{1c} and ϕ_{2c}, then the determinant of coefficients must vanish. The result of this condition is the operator equation:

$$(D_{1c}\nabla^2-\Sigma_{1c})(D_{2c}\nabla^2-\Sigma_{2c})-k\Sigma_{1c}\Sigma_{2c}=0 \qquad (3)$$

Knowing: $$L_c^2 = D_{2c}/\Sigma_{2c} \qquad (4)$$

and introducing the definitive relation

$$\Sigma_{1c} = D_{1c}/\tau_c \qquad (5)$$

equation (3) can be written as

$$\nabla^4-\left(\frac{1}{\tau_c}+\frac{1}{L_c^2}\right)\nabla^2-\frac{k-1}{\tau_c L_c^2}=0 \qquad (6)$$

where it is understood that the operators in equation (6) may operate on either ϕ_{1c} or ϕ_{2c}. In other words, the two fluxes must be represented by coupled general solutions of the same differential equation. Rewriting the above quadratic operator equation:

$$(\nabla^2+\mu^2)(\nabla^2-\nu^2)=0 \qquad (7)$$

where
$$\left.\begin{aligned}\mu^2&=\tfrac{1}{2}\left\{-\left(\frac{1}{\tau_c}+\frac{1}{L_c^2}\right)+\sqrt{\left(\frac{1}{\tau_c}+\frac{1}{L_c^2}\right)^2+\frac{4(k-1)}{\tau_c L_c^2}}\right\}\\ -\nu^2&=-\tfrac{1}{2}\left\{\left(\frac{1}{\tau_c}+\frac{1}{L_c^2}\right)+\sqrt{\left(\frac{1}{\tau_c}+\frac{1}{L_c^2}\right)^2+\frac{4(k-1)}{\tau_c L_c^2}}\right\}\end{aligned}\right\} \qquad (8)$$

where μ^2 and ν^2 are both positive, and μ^2 and $-\nu^2$ represent the positive and negative solutions of the LaPlacian, respectively. Thus the general solutions of (1) will involve linear combinations of μ^2 and $-\nu^2$, both parameters involving the properties of the core material. These, in turn, giving two pairs of equations, one pair of which is

$$\left.\begin{aligned} \nabla^2 \phi'_{1c} + \mu^2 \phi'_{1c} &= 0 \\ \nabla^2 \phi''_{1c} - \nu^2 \phi''_{1c} &= 0 \end{aligned}\right\} \qquad (9)$$

Each core flux is then a linear combination of the two values obtained from the separate solutions of equations (9) and the corresponding equations for ϕ_{2c}.

To determine the spacial dependence of flux in the axial (vertical) direction of the cylindrical core, solutions of (9) are obtained by separating variables (z and r) in the wave equation for the finite cylindrical reactor. This is facilitated by assuming the radial dependence of the flux is represented by a zero order Bessel function of the form $J_o(B_r\, r)$ where the parameter B_r is to be determined (B_r^2 representing effective radial buckling). Solutions of (9) may then be expressed as

$$\left.\begin{aligned} \phi'_{1c} &= A_z \cos \mu_z z \cdot J_o (B_r r) \\ \phi''_{1c} &= B_z \cosh \nu_z z \cdot J_o (B_r r) \end{aligned}\right\} \qquad (10)$$

where

$$\left.\begin{aligned} \mu_z^2 &= \mu^2 - B_r^2 \\ \nu_z^2 &= \nu^2 + B_r^2 \end{aligned}\right\} \qquad (11)$$

and similarly for ϕ'_{2c} and ϕ''_{2c} replacing constants A_z and B_z by A'_z and B'_z respectively.

Coupling coefficients are now introduced, i.e. $A'_z/A_z = S_1$ and $B'_z/B_z = S_2$, and the general solution of the system of equations becomes:

$$\left.\begin{aligned} \phi_{1c} &= \left[A\cos\mu_z z + B\cosh\nu_z z\right]\cdot J_0(B_r r) \\ \phi_{2c} &= \left[AS_1\cos\mu_z z + BS_2\cosh\nu_z z\right]\cdot J_0(B_r r) \end{aligned}\right\} \qquad (12)$$

Substitution of (12) into the least complex (i.e. the second) of the original steady state diffusion equations (1), shows that S_1 and S_2 are completely determined by the core constants obtained in the preliminary calculations.

Thus:

$$\left.\begin{aligned} S_1 &= \frac{D_{1c}L_c^2}{D_{2c}\tau_c}\left[\frac{1}{\mu^2\cdot L_c^2+1}\right] \\ S_2 &= -\frac{D_{1c}L_c^2}{D_{2c}\tau_c}\left[\frac{1}{\nu^2 L_c^2-1}\right] \end{aligned}\right\} \qquad (13)$$

The equations of continuity in the reflector region are slightly simplified due to the absence of the source term.

$$\left.\begin{aligned} D_{1R}\nabla^2\phi_{1R} - \Sigma_{1R}\phi_{1R} &= 0 \\ D_{2R}\nabla^2\phi_{2R} - \Sigma_{2R}\phi_{2R} + \Sigma_{1R}\phi_{1R} &= 0 \end{aligned}\right\} \qquad (14)$$

Noting that $$\Sigma_{1R} = D_{1R}/\tau_R \qquad (15)$$

and $$L_R^2 = D_{2R}/\Sigma_{2R} = D_R/\Sigma_R \qquad (16)$$

equations (14) may be rewritten as

$$\left.\begin{aligned} \nabla^2\phi_{1R} - 1/\tau_R\cdot\phi_{1R} &= 0 \\ \nabla^2\phi_{2R} - 1/L_R^2\cdot\phi_{2R} + \frac{D_{1R}}{D_R}\cdot\frac{1}{\tau_R}\cdot\phi_{1R} &= 0 \end{aligned}\right\} \qquad (17)$$

In a manner similar to that for the core region, the solution of equations (17) can be shown to be*

$$\left.\begin{aligned} \phi_{1R} &= C_z e^{-\mathcal{H}_{1z} z} \cdot J_o(B_r r) \\ \phi_{2R} &= S_3 \phi_{1R} + D_z e^{-\mathcal{H}_{2z} z} \cdot J_o(B_r r) \end{aligned}\right\} \qquad (18)$$

where

$$\left.\begin{aligned} \mathcal{H}_{1z} &= \sqrt{1/\tau_R + B_r^2} \gtrless 0 \quad \text{for } z \gtrless 0 \\ \mathcal{H}_{2z} &= \sqrt{1/L_R^2 + B_r^2} \gtrless 0 \quad \text{for } z \gtrless 0 \end{aligned}\right\} \qquad (19)$$

and S_3 is a third coupling coefficient expressible in terms of the pure reflector constants as

$$S_3 = \frac{D_{1R}}{D_{2R}} \left[\frac{L_R^2}{\tau_R - L_R^2} \right] \qquad (20)$$

The spatical dependence of the fast and thermal neutron flux in the core and reflector regions is expressed by equations (12) and (18), which involve 5 unknowns, the 4 constants A_z, B_z, C_z, and D_z, and the parameter B_r. The latter term may be specified simply by applying the boundary conditions at the core-reflector interface at the top of the cylinder (i.e. the condition of continuity of flux as well as that of the normal components of the neutron currents). Thus if the origin of coordinates is at the center of the cylinder, the interface in question is the plane $Z = H/2$ (where H was defined originally as the core height of 63.5 cm.) This leads to the following set of equations, in which the "critical determinant" of the coefficients of A_z, B_z, C_z, and D_z will vanish for a proper

*Abernathy (1).

choice of B_r.

$$\left.\begin{aligned}
&A_z \cos\mu_z \tfrac{H}{2} + B_z \cosh \nu_z \tfrac{H}{2} - C_z e^{-\varkappa_{1z}\frac{H}{2}} && (21) = 0\\
&A_z S_1 \cos\mu_z \tfrac{H}{2} + B_z S_2 \cosh\nu_z \tfrac{H}{2} - C_z S_3 e^{-\varkappa_{1z}\frac{H}{2}} - D_z e^{-\varkappa_{2z}\frac{H}{2}} && = 0\\
&-A_z \tfrac{D_{1M}}{D_{1R}} \mu_z \sin\mu_z \tfrac{H}{2} + B_z \tfrac{D_{1M}}{D_{1R}} \nu_z \sinh\nu_z \tfrac{H}{2} + C_z \varkappa_{1z} e^{-\varkappa_{1z}\frac{H}{2}} && = 0\\
&-A_z \tfrac{D_{2M}}{D_{1R}} S_1 \mu_z \sin\mu_z \tfrac{H}{2} + B_z \tfrac{D_{2M}}{D_{1R}} S_2 \nu_z \sinh\nu_z \tfrac{H}{2} + C_z S_3 \varkappa_{1z} e^{-\varkappa_{1z}H/2} + D_z \varkappa_{1z} e^{-\varkappa_{1z}H/2} && = 0
\end{aligned}\right\} (21)$$

For such a problem, a graphical solution is in order. This type of approach was employed by one author*, giving a value of B_r of 0.01250 cm^2. Since $\mu^2 = 0.01409\ cm^{-2}$, evaluated in terms of the core constants, then $\mu_z^2 = \mu^2 - B_r^2 = .00159\ cm^{-2}$. This quantity is equivalent to the effective axial buckling, and as such is indicative of the reflector savings. It can be shown** that the lowest eigen value of μ_z^2 is $(\pi/H)^2$. Thus, the effective height $H = \frac{\pi}{\mu_z} = 78.8$ cm, as compared to the actual height of 63.5 cm. Hence the reflector savings is 1/2 the difference, of 7.65 cm.

For the spatial dependence of flux in the radial (horizontal) direction, variables are again separated, now assuming axial dependence is described by $\cos\mu_z z$, where $\mu_z = .03987$, determined in the previous paragraphs. Using cylindrical functions, and imposing the same boundary conditions at the lateral core-reflector surface, a new critical determinant is obtained. The quantity R (radius of the critical clean core) is the parameter in this case, and the determinant is found to vanish (by graphical solution*) for the value R=14.8 cm.

Glasstone (pp214-215) shows the unreflected buckling

*See Abernathy (et al), pp213-214, (1).
**See, for example, Glasstone, Chapter VII

values for the finite cylinder and the rectangular parallelepiped. By equating these it is possible to determine the width of the critical rectangular core of height H. Thus,

$$B^2 = \left(\frac{2.4048}{R}\right)^2 + \left(\frac{\pi}{H}\right)^2 = 2\left(\frac{\pi}{W}\right)^2 + \left(\frac{\pi}{H}\right)^2 \qquad (22)$$

or $W = \frac{\sqrt{2}\,\pi R}{2.4048} = 27.34$ cm.

Critical Mass. The concentration of U^{235} was previously calculated to be 46.15 gms/ℓ. With dimensions of W= 27.34, and H = 63.5, the critical mass of the clean core is

$$M = 63.5 \times (27.34)^2 \times 46.15 \times 10^{-6} = 2.2 \text{ kg}$$

Critical Mass and Size of the Operating Core. Having calculated the lower limit of the critical mass of the water reflected clean core, a similar calculation assuming 10% burnup, fission product poisoning, and experimental losses will give a close approximation to the operating core critical mass. To do this it is necessary to express the reactivity losses, tabulated in Section 6 of this chapter (8.72% without burnup) as an equivalent uniformly distributed capture area in the core. The reactivity losses were calculated by making estimates of the core size (2.9 kg U^{235} and 61,500 cm^3). Thus, if the calculated criticality results are close to these, the reactor will satisfy the necessary operating requirements. Knowing the Xe^{135} reactivity loss is $\sim\Sigma = 0.0028\ cm^{-1}$ (uniformly distributed), a close approximation to the total loss is given by

$\Sigma_p = 8.72/3.12 \times (.0028) = .0078\ cm^{-1}$. The operating reactor must compensate for this loss when the fuel has been reduced by 10%. At such time, the new value of $\Sigma_u = .90 \times .0769 = .0692\ cm^{-1}$. Assuming transport properties unchanged, $\Sigma_M = .0195$, and the new values of z, L_M^2, k, and L_c^2 are

$$z = \frac{\Sigma_u}{\Sigma_M + \Sigma_p} = 2.535$$

$$L_M^2 = \frac{L_M^2 (original)}{1 + \Sigma_p/\Sigma_M} = \frac{8.92}{1.4} = 6.37\ cm^2$$

$$k = \eta z/1+z = 1.513$$

$$L_c^2 = \frac{L_M^2}{1+z} = \frac{6.37}{3.535} = 1.802\ cm^2$$

With the other constants remaining the same, the new calculations leads to an R = 16.98 cm, a corresponding W = 31.37 cm, a mass of 2.9 kg, and a volume of 61,500 cm^3. A reasonable estimate of the increase due to fuel plate removal would be 100 gms. Based on this, the critical mass becomes 3.0 kg.*

Critical Dimensions with Be-O Reflector. The same method of calculation (with greatly increased complexity due to the presence of three regions instead of two), leads to a critical mass of 1.16 kg of U^{235} and a core width of 19.4 cm.**

Further Theoretical Considerations. In the type of calculations discussed thus far, certain simplifying assumptions were made in order to preserve elementary methods of treatment. When refinements must be made to determine small compensatory changes in reactivity it is necessary to employ multi-region

*See pp 221-222, AECD 3557 (1)
** See p 237 (1)

reflector calculations and perturbation theory. For example, the effect on criticality of removal of fuel plates for placement of control and safety rods in the core is most easily approached by perturbation theory. This and other considerations involving changes in reactor parameters in localized portions of the core and reflector are treated in the unclassified literature.*

2. Flux Data.

At one megawatt operation the average thermal flux ranges from the 7.61×10^{12} of the clean core to 8.45×10^{12} for the operating core (at 10% burnup). Since the flux varies inversely with the critical mass, the use of the Be-O reflector almost doubles the average thermal flux to a value of $\backsim 1.2 \times 10^{13}$. The spatial variation of fast and slow flux in the core and reflector can be obtained from equations (12) and (18). No attempt is made herein to verify these calculations or to indicate the effects of changes in geometry. Suffice to say that these matters also involve detailed mathematical procedures, based on the application of first order perturbation theory.

3. Thermal Utilization Factor.

In the preceding calculations it was assumed that the reactor core consisted on a homogeneous mixture of water, aluminum, and U^{235}. The corresponding thermal utilization factor is $f = k/\eta = 1.685/2.11 = 0.8$ (the fraction of the total thermal

*Treated generally by Erlich (10) and Glasstone (16), with some detailed calculations for the low cost reactor presented by Abernathy (1), pp 81-103.

neutrons absorbed in fuel material). for a clean, unpoisoned core. To check the validity of this assumption, an improved calculation can be made, using a model in which the fuel is distributed uniformly over the entire fuel plate, but fuel and moderator water are separated.* The resulting value of f is 0.81, and the local flux depression in a fuel plate is revealed to be on the order of a few percent at most. Thus, the homogeneous model is within the permissable limits of error in calculations pertaining to the pool reactor core.

4. Temperature Coefficient.

Reactor operation involves temperature variations which effect the reactivity. These effects are primarily due to an increase in the mean energy of thermal neutrons with increasing temperature (with a consequent variation in nuclear cross-sections), and changes in the mean free paths and the non-leakage probabilities due to the density changes. Thus the temperature is a combination of both nuclear and density factors. The former is related to the $1/v$ (or $1/\sqrt{T}$) dependence of microscopic absorption cross-sections, and the latter is manifest in the variation of the effective multiplication factor with changes in core geometry and material densities. Satisfactory reactor control requires a small negative temperature coefficient, with the steady state maintained by control rods. The primary reason for stability with a negative coefficient is the increase in Fermi age and diffusion length as the moderator density decreases, resulting in increased neutron leakage

*See page 62, reference (1)

from the core.

The temperature coefficient is given by the logarithmic derivative (with respect to T) of the effective multiplication factor. Thus, since

$$K_{eff} = \frac{\eta f}{(1+\tau_M B^2)(1+L_c^2 B^2)}$$

$$\frac{1}{K_{eff}} \cdot \frac{\delta K_{eff}}{\delta T} = \frac{1}{\eta}\frac{\delta \eta}{\delta T} + \frac{1}{f}\frac{\delta f}{\delta T} - \left(\frac{\tau_M}{1+\tau_M B^2} + \frac{L_c^2}{1+L_c^2 B^2}\right)\frac{\delta B^2}{\delta T} - \left(\frac{B^2}{1+\tau_M B^2}\right)\frac{\delta \tau_M}{\delta T} - \left(\frac{B^2}{1+L_c^2 B^2}\right)\frac{\delta L_c^2}{\delta T}$$

Substitution of appropriate values in the above equation shows a .24% decrease in reactor reactivity for a mean temperature increase in the system of 20 degrees C. This is equivalent to the -.0075%/degree F cited in the literature for the reactor temperature coefficient.

5. Fission Product Poisoning.

With the operation of a high neutron flux reactor, the fission process results in the poisoning of the core by neutron absorbing fission products. Some of these are of short half-life (hence will reach equilibrium concentration which is a function of the reactor operating power level), while some are long half-life whose concentrations grow steadily with continued operation. The effects of the latter type poison can be estimated in terms of operating level and time (giving total # of fissions) and average capture cross-sections ($\sim$ 50 barns). For example, using constants of Table V and supposing 300 gms of fuel have been destroyed (i.e. 10% burnup), then the mass fissioned is

$$\frac{300}{1+\left(\frac{\sigma_c}{\sigma_f}\right)_{U^{235}}} = \frac{300}{1.184} = 253 \text{ gms.}$$

The total number of fissions is

$$\frac{253 \times 0.602 \times 10^{24}}{235} = 6.47 \times 10^{23}$$

Thus the total capture area (50 barns/fission = 50 x 10^{-24} cm^2/fission) is 32.4 cm^2 for a core volume of 61,500 cm^3. Since poisoning, P, is defined as the ratio of thermal neutrons absorbed by poison to those absorbed in fuel, cumulative small cross-section fission product poisoning is

$$P = \frac{\Sigma_{low\ \sigma_a}}{\Sigma_u} = \frac{32.4/61{,}500}{.077} = .00748$$

and excess reactivity cost is

$$\rho_p = \frac{.00748}{1+\Sigma_M/\Sigma_u} = \frac{.00748}{1.28} = .0057 = .57\%$$

The effects of short half-life poisons such as Xe^{135} (σ_c = 3.5 x 10^6 barns) must be calculated on the basis of the average thermal flux during operation. Knowing that

$$\eta \Sigma_u \bar{\phi}_{th} = \nu \Sigma_f \bar{\phi}_{th} = \frac{\nu \times 3.1 \times 10^{13} \cdot P\,(Kw)}{Vol\ core\ (cm^3)}$$

or, using values previously tabulated, the average thermal flux is:

$$\bar{\phi}_{th} = \frac{2.2 \times 10^{10} P\,(kw)}{M_u\,(kg)} = 8.45 \times 10^{12}\ cm^{-2} sec^{-1}$$

for P = 1 megawatt, and M_u = 2.6 kg (2.9 kg fuel mass reduced by 10% burnup). Following the method of Glasstone (pp329-339) and (1), using data consistent with previous calculations, and assuming uniform distribution of fission products (a reasonable assumption considering the combined effects of control rods, reflector, etc., in flattening the neutron flux distribution),

the steady state poisoning, P_{ssXe}, due to Xe^{135} is

$$P_{ssXe} = \frac{\Sigma_{Xe}}{\Sigma_u} = \frac{\sigma_2 (\gamma_1+\gamma_2)\Sigma_f \bar{\phi}_{th}}{(\lambda_2 + \sigma_2 \bar{\phi}_{th})\Sigma_u}$$

where $\gamma = .061 = I^{135}$ direct fission yield, and $\gamma_2 = .003 =$ Xe^{135} direct fission yield. The corresponding reactivity loss due to Xe^{135} in equilibrium is $\left(\frac{\delta Keff}{Keff}\right)_{ssXe} = \frac{P_{ssXe}}{1+\Sigma_M/\Sigma_u}$

The maximum Xe^{135} poisoning is given by:

$$P_{max} = P_{ssXe}\left[e^{-\lambda_2 t_m} + \left(\frac{\gamma_1}{\gamma_1+\gamma_2}\right)\left(\frac{\lambda_2+\sigma_2\bar{\phi}_{th}}{\lambda_2-\lambda_1}\right)\left(e^{-\lambda_1 t_m} - e^{-\lambda_2 t_m}\right)\right]$$

at time $t_m = -ln\left\{\frac{\lambda_2}{\lambda_1}\left[1+\left(\frac{\gamma_1+\gamma_2}{\gamma_1}\right)\left(\frac{\lambda_1-\lambda_2}{\lambda_2+\sigma_2\bar{\phi}_{th}}\right)\right]\right\}$

with a reactivity loss of

$$\left(\frac{\delta Keff}{Keff}\right)_{max} = \frac{P_{Xe(max)}}{1+\Sigma_M/\Sigma_u}$$

Using $\gamma_1 = .061$, $\gamma_2 = .003$, $\lambda_1 = 2.9 \times 10^{-5}$ sec^{-1}, $\lambda_2 = 2.1 \times 10^{-5}$, $\sigma_{Xe} = 3.5 \times 10^6 = \sigma_2$, $\Sigma_{Xe} = .0028$ cm^{-1}, $\bar{\phi}_{th} = 8.45 \times 10^{12}$, $P_{ssXe} = .032$, $t_m = 2.11 \times 10^4$ sec ≈ 5.9 hrs., and $P_{Xe(max)}$ $\approx .04$. With 10% burnup, $\Sigma_u = .0692$ and $\Sigma_M = .0195$. Thus, for 1 megawatt operation, excess reactivity loss can be calculated to be ~2.5% reaching 3.12% due to max concentration of Xe^{135} after shutdown. Inspection of the poisoning equation reveals that the poisoning is relatively negligible for values of $\bar{\phi}_{th}$ of 10^{11} or less, but for values approaching 10^{13} and greater, the poisoning increases rapidly, to a limiting maximamum at about 10^{15}.

6. Excess Reactivity Requirements.

Employing methods similar to those of the previous two sections the excess reactivity requirements can be estimated for effects of temperature change, burnup, fission product poisoning, and experimental absorption. This has been done for 1 megawatt operation (1), and 100 kw operation (8). The results are tabulated below for comparison.

TABLE VI

ESTIMATED REACTIVITY ALLOWANCES (%ρ)

100 kw		Allowance For:		1000 kw
.375	(30°F)	Temperature Change	(20°C)	0.24
.45		Xenon Poisoning		3.12
.15		Low σ_c Fission Poisons		.57
		Burnup (10%)		2.20
2.00		Experimental Absorption		2.75
		Beam Holes (3 of 6" diameter)		2.04
.30		Operational Control Requirements		
3.275		Total		10.92

7. Heat Transfer and Cooling System.

An analysis of the heat transfer for the 1000 kw forced cooling system, and the 100 kw convection cooling is carried out in detail by Berger (4). The results reveal the following: for the 1000 kw operating level, forced circulation cooling (2 ft/sec flow velocity between fuel plates): 1) temperature rise of coolant water between entrance and outlet is 9.25° F; 2)tem-

perature of fuel plate walls, 181.9° F, center of fuel plates 183.1° F; 3) power level at which boiling first takes place is 1590 kw; 4) pumping rate required, 774 gal/min, and pressure drop throughout system 19.44 psi.; 5) the heat exchanger requirements include entrance temperature for heated water 88°F, cooling water entrance temperature 65°F, and heat exchanger exit temperature 80°F.

Note that the 2 ft/sec flow velocity (corresponding Reynolds Number 15,300) is a requirement to provide adequate margin of safety in maintaining fuel plate temperature below the boiling point of water (239°F at the given depth). Moreover, the calculations readily verify that thermal stress is not a problem in this type of fuel plate. The recommended pump for the system is a 1250 gal/min, 50 ft head pump in order to allow for overload and possible hot spots overlooked in the calculations. Based on available heat exchange data, an exchanger 20 ft long with a 19 in. diameter shell, U-tube, costing an estimated $3400, will satisfy requirements of the system.

A solution to the problem of convective cooling for 100kw operation presents considerably more difficulty. Employing a method set forth by Schwartz* the following results are obtained:** temperature rise across reactor 11°F, and flow velocity between fuel plates .156 ft/sec. Applying empirically derived formulae it is revealed that providing pool purging of ~ 65

* Schwartz, H., Natural Convection Cooling of Liquid Homogeneous Reactors. AECU-706, ORNL, 1951.
**Berger (4)

gallons/min, an equilibrium pool temperature of 80°F can be maintained, and corresponding reactor wall temperatures will be ~133°F to 146°F (depending on the value of empirically derived heat transfer coefficients), a figure well below the boiling point.

Trial and error substitutions into the equation defining the heat transfer coefficient, $h_p = 38.5\ (\Delta T)^{\frac{1}{4}}_p$, (with $\Delta T = 41.7^{\circ}F$ and $h_p = 97$ BTU/ hr.ft^2-°F at 100kw), reveals the 400 kw power level as that at which boiling may be anticipated.** This is consistent with the "optimistic limiting case" in the numerical analysis made by Claiborne and Poppendick of ORNL, (6). The analysis of limiting cases was necessary due to the number of unknowns in the existing knowledge of transient boiling heat transfer, and bubble formation. Thus, one case assumes boiling as soon as the wall-coolant temperature reaches the boiling temperature. Steam bubbles form, and after .127 seconds, 10% steam has made the reactor sub-critical. The opposite case, in which boiling occurs but bubble formation is delayed by subcooling, indicates that in .14 seconds the fuel plates reach melting temperature. Although it was mentioned previously that the condition of prompt critical is extremely unlikely, a more thorough analysis of the transient thermal behavior, if desired, must await the accumulation of fundamental information on transient boiling heat transfer.

8. Shielding Data.

General. The shielding thicknesses mentioned in Chapter II

**Berger (4)

were calculated on the basis of data from the Oak Ridge Bulk Shielding Reactor, where gamma and neutron fluxes were measured on the centerline at various distances from the reactor, employing the equation:

$$D(x) = \frac{S\mu x e^{-\mu x}}{4\pi x^2}$$

where D = dose rate, x = shield thickness, μ = absorption coefficient, S = source strength, and using a μ = .0285 cm^{-1} corresponding to 5.5 mev gammas. This gave the very conservative value of 21.2 ft for required depth of the reactor in water, and 8 ft of concrete plus 4 ft of water to provide shielding to 1/10 tolerance dose rate. (Note, for concrete, density 2.33 g/cc, μ = 0.0667 cm^{-1}, thus 2.13 ft of water 1 ft of water $\approx$ 1 ft of concrete). These are centerline requirements, hence proportionately less concrete is required, above the centerline.

Activity in the walls which will produce a tolerance dose rate at the surface has been calculated by estimating flux from data on neutron activation (Leddicotte, (24)). However, a thickness of 4 ft of water reduces all activation to well below tolerance limits. Scattering is not a shielding problem since all paths of escape offer satisfactory attenuation.

Activity in the water arises from activation by neutron capture, and from aluminum atoms recoiling into the water after neutron capture. The activity of any element is proportional to the product of its concentration and absorption cross-section per gram. With no purge, saturated activity is inde-

pendent of half-life, but pool purging will easily eliminate long half-life activity. By using the following expressions, the dose-rate from recoil activity, and the allowable concentration of elements in the water can be calculated.

Long half-life activities:

$$A = \frac{\frac{.6023}{4a} \cdot SR\rho\sigma\bar{\phi}}{V_p + P/\lambda} \cdot \left(1 - e^{-(\lambda + P/V_p)t}\right)$$

Short half-life activities:

$$A = \frac{.6023}{4aF} \cdot \lambda \cdot SR\rho\sigma\bar{\phi}\, e^{-\lambda t}$$

where A = activity in disintegrations/unit volume/sec.

λ = decay constant of active atom

F = flow rate of cooling water through reactor core

t = time of decay

a = atomic weight of element

S = surface area of Al in active lattice of core

R = range of recoil nucleus

ρ = density of aluminum

σ = activation cross-section/atom

$\bar{\phi}$ = average thermal flux

P = purge rate

V_p = volume of water in pool

Concerning aluminum activation, there are several possible reactions.

For [illegible]:

$$Al^{27} + n \longrightarrow Na^{24} + \alpha$$

$$Al^{27} + n \longrightarrow Mg^{27} + p$$

$$Al^{27} + n \longrightarrow Al^{28} + \gamma$$

Slow neutrons: $Al^{27} + n \longrightarrow Al^{28} + \gamma$

Na^{24} recoils give only 1% of a tolerance dose rate at the surface, and the others are even more negligible, decaying near the bottom of the pool.

In analyzing the activation of water, it is learned that sodium (half-life 14.9 hrs) gives rise to the largest source of activity and really the only one worth consideration. Since the sodium content of water varies depending on its source, the pool water may have to be demineralized or distilled. However, since only the activity on the surface contributes to the dose-rate, a likely solution to the problem is a small surface purge ($\sim$10 gpm) which would keep a fresh and inactive layer of water on the surface as a shield.

Since the maximum permissable concentration of Na^{24} in drinking water is 300 disintegrations/sec/cc, pool water with its maximum on the order of 100 disint/sec/cc can be discharged directly.

The tolerance rate activity is a function of both the number and energy of gammas given off. While the gamma from thermal neutron activation of most of the common elements has an energy range of 1.1 mev to 2.76 mev (with most lying closer to 1.1 mev), the gammas which are of concern after shutdown are of energies above 2 mev. A satisfactory value for computation is 3 mev, which, by means of the following relation

$$\mu r \frac{e^{-\mu r}}{r^2} = \frac{4\pi I_0}{F \sum_r^{\infty} A_\gamma E e^{-\lambda t}}$$

where: EI_0 = 5250 mev/sec/sq cm

$F = 3.1 \times 10^{16}$ fissions /sec @ 1000 kw

A_γ = Gammas per fission at saturation

r = distance from source

μ = absorption coefficient

shows that the shield of water, required after various times of decay ranges from 15.5 ft at 10 sec to 10.0 ft in 1 day.* Thus 11 ft was chosen as a reasonable value to permit access to the pool a few hours after shutdown. This 11 ft refers to the shielding layer of water in the "storage" end of the pool, where the hot fuel elements are stored in a rack after shutdown.

Experimental data on induced activity in the 58 kg of aluminum (2S Al) of the reactor structure shows that only 3 ft of water is a more than adequate shield.* Similarly, calculations concerning the shielding of the heat exchanger reveals that the 6 ft water shield provided by the reactor design is again more than adequate.

Shielding of beam holes is easily accomplished by liners, movable concrete blocks, and lead coffins for the plugs. In general, this problem is one which is easily solved by the experimenter to meet the demands of the experimentation being

* See pp 167-170, reference (1).

performed.

The problems of shielding at 100 kw are much less. With convective cooling, flow rate is decreased, and short half-life activity in the water becomes negligible. Moreover, the allowable sodium concentration can be raised by a factor of ten, the water shield depth reduced to 16.5 ft, and concrete shield thickness reduced to $\backsim$3 ft.

BIBLIOGRAPHY

1. Abernathy, F. H. (et al). A Low Cost Experimental Neutron Chain Reactor. AECD 3557, ORNL, November 1952.

2. Anderson, E. E., (et al). A Laboratory Course in Reactor Physics. AECU - 2164, ORNL, August 1952.

3. Beck, C. K. Uses and Limitations of a Low-Power Reactor. Nucleonics, 9:18-27, November 1951.

4. Berger, W. A., Heat Transfer Problems in a Low Cost Nuclear Reactor. ORSORT, 1952.

5. Breazeale, W. M. The "Swimming Pool" - A Low Cost Research Reactor. Nucleonics, 10:56-60, November 1952.

6. Breazeale, W. M. A Low Cost Experimental Neutron Chain Reactor. AECD-3435 (ORNL 1105 rev), April 1952.

7. Chernick, J. The Dependence of Reactor Kinetics on Temperature. BNL-173, December 1951.

8. Cole, T. E. Design of a Control System for a Low Cost Research Reactor. Nucleonics. 11:32-37, February 1953.

9. Dwyer, O. E., (et al). Heat Transfer Rates for Cross-flow of Water Through a Tube Bank at Reynolds Numbers up to a Million. Conference on Nuclear Engineering, University of California. F-73 to F-90. Berkeley, 1953.

10. Erlich, R., and Hurwitz, H. Multigroup Methods for Neutron Diffusion Problems. Nucleonics. 12:23-30, Feb. 1954.

11. Emmons, A. H., and Lauderdale, R. A. A Method for Estimating Low Levels of Radioactive Contaminants in Water. Nucleonics. 10:22-23, June 1952.

12. Evans, G. E. Reactor Materials. Nucleonics. 11:18-26, June 1953.

13. Fermi, E. Nuclear Physics, University of Chicago Press, 1950.

14. Fox, M. Uses of a Reactor for Research. Nucleonics. 11:46-48, June 1953.

15. Franz, J. P. Pile Transfer Functions. AECD-3260, ORNL, July 1949.

16. Glasstone, S., and Edlund, M. C. The Elements of Nuclear Reactor Theory. D. Van Nostrand, New York, 1952.

17. Goodman, C., ed. The Science and Engineering of Nuclear Power. Vol. I, Introduction to Pile Theory. Vol. II, Applications of Nuclear Energy. Addison Wesley Press, Cambridge, Mass., 1947.

18. Harper, J. M. Reactor Operation. Nucleonics. 11:35-40, June 1953.

19. Hughes, D. J. Pile Nuetron Research. Addison Wesley, Cambridge, Mass., 1953.

20. Hughes, D. J. The Reactor as a Neutron Source. Nucleonics. 11:30-35, January 1953.

21. Hurwitz, H. Derivation and Integration of the Pile-Kinetic Equations. Nucleonics. 5:61-67, July 1949.

22. Isbin, H. S. Nuclear Reactor Catalog. Nucleonics. 11:65-69, June 1953.

23. Lansing, N. F. The Role of Engineering in Nuclear Energy Development. TID-5031. ORNL, December 1951.

24. Leddicotte, G. W., and Reynolds, S. A. Activation Analysis with the Oak Ridge Reactor. Nucleonics. 8:62,Mar. 1951.

25. Morgan, K. Z., (et al). Health Physics Insurance Seminar. TID-388. ORNL, March 1951.

26. National Bureau of Standards, Handbook 47. Recommendations of the International Commission of Radiological Protection. U.S. Dept. of Commerce. June 1951.

27. National Bureau of Standards, Handbook 48. Control and Removal of Radioactive Contamination in Laboratories. U.S. Dept. of Commerce. December 1951.

28. Nuclear Engineering, 1953 Conference Proceedings. University of California, September 1953.

29. Problems in the Use of Fission Products. Nucleonics. 10:45-47, January 1952.

30. Smith, J. M. Radiation Safety for a Reactor. Nucleonics. 11:41-45, June 1953.

31. Soodak, H., and Campbell, E. C. Elementary Pile Theory. John Wiley and Sons, New York, 1950.

32. Weinberg, A. M. Pile Neutron Physics. MDDC-1720, ORNL, February 1948.

33. Weinberg, A. M., and Noderer, L. C. Theory of Neutron Chain Reactions. AECD-3471, ORNL, May 1951.

34. Woodruff, O. J., (et al). Reactor Coolants. Nucleonics. 11:27-34, June 1953.