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THE MIGHTY MOUSE RESEARCH REACTOR
PRELIMINARY DESIGN STUDY

by

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ABSTRACT

The design of a 250-mw heterogeneous reactor, with a peak thermal neutron flux level of about $5 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$, is proposed for basic research, neutron-beam experimentation, and irradiation studies at an Argonne site.

The design concept features a D_2O -cooled, moderated, and reflected annular core (30 cm ID, 90 cm OD, 90 cm high) of highly enriched, magnesium-clad fuel elements contained in a pressure vessel approximately 13 ft in diameter.

Included are the results of preliminary studies directed toward establishing the avenues of research and development necessary to resolve the problems of heat transfer, fuel fabrication and handling, control, and shielding inherent to the proposed reactor operating cycle of 5 days.

I. INTRODUCTION

This report is prepared as the first-stage conceptual design and feasibility study of a high-flux reactor for basic research as opposed to one for engineering tests. When the neutron flux reaches the level used in the proposed reactor [the peak of the unperturbed thermal flux is $\sim 5 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$], it is necessary to remove a large quantity of heat from a relatively small volume of core. Thus the reactor name assigned is that of a characterization having a large quantity of muscle contained in a small body - Mighty Mouse.

The design of reactors for experiments almost always involves a compromise between two extremes. The first extreme is a reactor having a versatility of size of experimental facilities and of irradiation flux levels. The other extreme is a reactor designed for a single user and a single experiment. The latter, while ideal for the experimenter, is too costly an approach for general use.

The test reactor category refers to a reactor capable of handling irradiations of reactor components. The predominant components are reactor fuel elements or assemblies which usually demand large irradiation holes with a fairly uniform neutron flux over a large volume. Consequently, the volume of the reactor core must be large. Further, if it is necessary or desirable to have an average thermal neutron flux of $10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ inside an irradiation hole when the experiment is in place, the reactor power level is almost certainly 500mw, and may well be over 1000 mw if a number of major test facilities are to be provided.

A reactor designed to meet the demands of basic research may have a relatively small core and small irradiation facilities. Such a research reactor can furnish an average thermal neutron flux of over $10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ in a small volume inside an irradiation facility and operate at a power level under 500 mw.

While it is possible to design a reactor which combines component and basic research irradiation facilities, the two types are not particularly compatible. They can and will interfere with each other. The flux would either be excessively perturbed in the area of the component irradiation facility, or the experiment would change the neutron and gamma spectrum to a degree which would invalidate any research experiment in an adjacent test facility. In the long run, the reactor is operated predominantly for one type of experiment or the other. Consequently, some of the utility of such a reactor is lost.

The highest priority for irradiations is given to experiments of a nature requiring a test-type reactor. As a consequence, most high-flux space in existing reactors is occupied by component test facilities. Most component tests (excluding possible military tests) cannot justify a thermal flux of $10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ for its own sake. This flux level is usually justified on the basis of shortening irradiation time. These demands might better be met by building, for example, more reactors having fluxes of $10^{14} \text{ n}/(\text{cm}^2)(\text{sec})$. The users who can justify the higher flux level, or who find it essential to shorten irradiation time, frequently require highly specialized irradiation facilities. This demand can be best met by highly specialized reactors.

In view of the high demand for test-type reactors, they appear to be of a type more readily built by private industry for their own use and for space rental to others. This leaves the development program for high-flux research reactors to the National Laboratories.

The best of the high-flux research reactors in operation is the MTR, designed primarily for basic research. However, the demand for materials testing is great enough that the MTR reactor usage is predominantly for test work. Little or no "clean" reactor test volume is left for research experiments. The high priority given component irradiations has left a large, unanswered demand for high-flux facilities for basic research.

Since test reactor and research reactor programs are not especially compatible in a single reactor, and since it is a major problem to design either type of reactor with just a reasonable degree of versatility, the Mighty Mouse reactor is designed for, and is intended to be used largely by, those conducting basic research studies.

The greatest and most economic utilization of such a reactor is realized when the reactor is located at the site of, and operated by, the major user. Thus the planning calls for construction at the Argonne National Laboratory. While it is not claimed that Argonne will be the sole user of this reactor, the Laboratory would be the major user.

The need and justification for this reactor are found in research uses which fall into two major categories: irradiations and beam experiments.

The irradiation experiments involve such items as:

- (1) study of damage in fissionable material as a function of fission rate;
- (2) activation of materials having low absorption cross sections;
- (3) production of materials having short half-lives;
- (4) production of materials requiring multiple-order capture products, especially where intervening half-lives are short;
- (5) studies on isotopes of small abundance;
- (6) activation analyses of small samples or small concentrations of impurities in samples;
- (7) chemical studies, such as radical-radical interactions and steady-state concentrations of H_2 and H_2O_2 or D_2 and D_2O_2 in water;
- (8) activation of materials to higher levels of specific activity.

The beam experiments are, for the most part, a utilization of the high-intensity neutron flux to obtain either better resolution of measurements or to extend the currently established techniques of determining physical constants into regions not possible with the neutron fluxes now available. The following are examples of beam experiments which can utilize high flux:

- (1) neutron diffraction experiments on crystals of low symmetry and on both small powdered samples and small single crystals;

- (2) extension of work using neutron choppers;
- (3) improved resolution of experiments with neutrons polarized by scattering;
- (4) improved resolution of measurements using crystal and beta spectrometers.

The proposed reactor could, with the type of fuel elements designed for it, be operated to produce isotopes within the core. The most probable mechanism is to substitute producer plates or tubes, identical in geometry with fuel plates or tubes, for some of the fuel elements within the assemblies. Such producer elements may be loaded with a quantity of production material such that the unit heat release during reactor operation is the same as for a fuel element.

While it is obvious that the high flux level in the Mighty Mouse reactor can produce material having a higher specific activity than in any existing reactor, it is well to point out the value of one application. This reactor could, by long-term irradiations starting with uranium or plutonium, produce quantities of californium, which would in itself be a valuable research material. The high rate of spontaneous fission of californium makes it an excellent point source of neutrons for work such as surveying a reactor core to determine the worth of various regions of the core.

The design for such a reactor started at Argonne in the late spring of 1956 and proceeded somewhat intermittently as manpower was available. The effort to date has involved studies of the reactor to determine fuel requirements, flux patterns, and some of the control requirements. The heat transfer and design studies have included the development of various fuel designs and evaluation of the characteristics of pressure drop and heat transfer. Thought has been given to the general direction for the development program of the fuel-handling system, the control system, and the reactor tank design. Only preliminary consideration has been given to the size of facilities, such as for the reactor building and water plants. However, data are available for making first-stage approximations.

The studies thus far have dealt with those major problems which, if they appear to have a reasonable solution, make the establishment of a development program for the reactor a worthwhile venture. The physics of the reactor, the heat transfer conditions in the core, and the design difficulty of various components, e.g., fuel handling and control, appear to be not just amenable to a satisfactory solution, but require only relatively small extrapolations from current reactor practice. While no claim is made that solutions are easy, it does not appear that a great variety of new and novel mechanisms must be devised, that especially exacting heat transfer conditions will be found, or that control problems are an order of magnitude higher than in existing reactors.

The data listed in Table I represents a goal for an advanced type of research reactor. The data are the result of preliminary calculations; however, they are proof that reaching a goal of a tenfold increase in thermal neutron flux is a reasonable extrapolation from current practice.

Table I

REACTOR DATA

Reactor Power Level	250 mw
Reactor Operating Pressure	~ 60 psia
Fuel	U^{235} in Mg
Fuel Loading	~ 3.3 kg U^{235}
Fuel Operating Cycle	~ 120 hr
Reflector	~ 5 ft D_2O
Maximum Fast Flux	~ 2×10^{15} n/(cm ²)(sec)
Maximum Thermal Flux	~ 5×10^{15} n/(cm ²)(sec)
Moderator and Coolant	D_2O
D_2O Flow Rate to Core	~ 45,000 gpm
D_2O Inlet Temperature	140F
D_2O Exit Temperature	175F
Average Heat Flux	475,000 Btu/(hr)(ft ²)
Maximum-to-average Heat Flux Ratio	1.34

II. DESIGN CONCEPTS

A. Reactor

1. Core

The active section of the core is contained in a hollow cylinder, 90 cm outside diameter, 30 cm inside diameter and 90 cm high. Within these physical limitations, studies of fuel design have been made using four primary considerations:

- (1) The fuel elements should be of a simple design and economical to manufacture, because of the large quantity required by the short lifetime of the core.
- (2) The core design must be such that adverse design problems are not encountered with experimental facilities.
- (3) The solution of the transient heat removal problems of coast-down and scram of this high-power system must be satisfactory.
- (4) The burnout limits must be clearly established.

Three designs of fuel element appear to have particular merit for the proposed reactor. These elements, shown in Figs. 1 to 3, are an axial flow, tubular element design, and two axial flow, plate element designs. The tubular geometry is preferred since the elements appear to be more easily fabricated and more readily assembled into final form. However, the final choice of fuel element and assembly will be dependent upon a compromise among considerations of physics, heat transfer, fluid flow, metallurgy and, possibly, processing.

All fuel assemblies studied use magnesium as the clad metal on the uranium or the uranium-magnesium matrix, and as the structural metal within and directly adjacent to the active zone. In the past, magnesium has been given very little consideration as a material to be in contact with water in a reactor system, since the corrosion rate is assumed to be intolerably high. However, laboratory tests at Argonne^{1,2} indicate that, even under somewhat adverse conditions for magnesium, the corrosion rates are tolerable for a 5-day operating cycle. Under more optimum conditions (pH~9.5), corrosion rates involving less than 1 mil/mo penetration were observed in water at 25 fps and at temperatures ranging from 160-170F.

¹Private communication, J. E. Draley to S. Untermeyer (September 17, 1952)

²W. Seefeldt, not published.

The uranium in the fuel elements is contained as a layer of uranium metal or uranium compound in the center of the fuel plates or tubes, or as a dispersion of metal or compound within a center zone of the tubes or plates. In all cases, the thickness of the fabricated fuel is about 30 mils. The fuel assemblies are relatively large, being illustrated as occupying approximately a 41-degree segment of the annular core. Large fuel elements are desired in order to minimize the number of pieces handled during the fuel loading and unloading period.

The coolant enters at the bottom of the core and leaves at the top. The bottom supporting grid is divided into 8 radial segments of 45 degrees each, approximately 41 degrees is assigned to fuel and 4 degrees to control. The tubular design (Fig. 1) has coolant both internally and externally. The final assembly is a welded structure. It is hoped, however, that no welding will be required on the respective fuel tubes. In the plate designs the 30-mil plates are spaced on 180-mil centers. The fuel plate or strip is made preferably in 2 to 3-inch widths. The plate types shown illustrate the possible variations in designs which, for the design shown in Fig. 2, require furnace brazing the fuel plates in the assembly or, for the design shown in Fig. 3, require spring pressure of curved plates to obtain a sufficiently rigid, vibration-free structure.

The fuel assemblies are to be supported at the top with a locking grid which may be anchored to a structure within the reactor vessel or to the top plug containing the control rod drive systems.

2. Control

A major portion of the total control is handled as burnup control. The preferred method is one in which a poison or a combination of poisons, such as lithium, boron and cadmium, are incorporated in the fuel elements. The burnup characteristics of the incorporated poisons are made to match that of the U^{235} as closely as possible.

An alternate method of the control with burnable poisons is to incorporate them into the D_2O coolant in some soluble form. The concentration of the soluble poisons could be adjusted continuously, either by adding fresh solution or removing the soluble material with ion exchange resins. The control method by use of soluble poison is most applicable to the alternate reactor vessel design (see page 32) where the radial moderator-reflector zone is a separate circulating system which would contain no poison. However, regardless of the reactor design, the incorporation of the burnable poison into the fuel elements is much preferred to the use of soluble poisons. The latter would be considered only if it were impossible to match burnup characteristics or if metallurgical difficulties were encountered.

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Since the high neutron flux causes a degree of burnup of poison in a normal type of shim rod such that frequent rod recalibration is necessary, the shim control proposed is a network of two or three parallel and separate systems of circulating BF_3 . The tubes containing the BF_3 would penetrate the full axial dimension of the core. The tube would be chosen such that, within the pressure limits between atmospheric and 10 to 15 atmospheres, both shim control or shutdown can be accomplished. Calibration of the BF_3 systems is maintained by continuously purifying the BF_3 and by accurate control of atom density within the active zone. Four of the eight radial control slots in the core are assigned to the circulating BF_3 system. A helium purge system is available for removing BF_3 or for drying the systems. Although the BF_3 system is intended for shim control, it is possible to use this for shutdown by dumping BF_3 into the system up to the full working pressure of the system.

Two of the remaining four control slots are assigned to regulating control. These are blades of stainless steel. Any of the standard Type 300 series stainless steels have an adequate cross section (833 gm of stainless steel $\sim$ 1% k). Due to the relatively low cross section of this regulating material, burnup is slow and the recalibration of blades should not be difficult. The regulating blades occupy positions 180° apart. The drives, scram mechanisms, and shock absorbers have not been studied. It is expected that conventional designs can be used and that no unusual specifications for drive speeds or scram times should be evolved.

The remaining two control slots are assigned to safety blades. Boron-stainless steel blades are considered adequate. These blades, as well as the regulating rod blades, disconnect from their drives and remain in their core positions during fuel loading and unloading operations.

3. Pressure Vessel

The reactor pressure vessel (Fig. 4) is 13 ft in diameter. The over-all vessel height is to be determined by a combination of desired axial reflector plus structural requirements for the core, the control systems, the coolant lines and the experimental facilities. The estimated height is 25 ft. The most probable material of construction is one of the Type 300 series stainless steels. However, since the primary coolant may be maintained in an alkaline condition, the environment may be quite good for acceptably low corrosion rates on lower cost, more easily fabricated steels. The vessel is designed for a working pressure of 100 psi, although the calculations indicate that 60 psia pressure at the top of the core is adequate.

If the main body of the vessel is stainless steel or any other ferrous alloy, it is desired that the section adjacent to the thermal column (Figs. 5 and 6) be a material of lower neutron absorption cross section, such as zirconium. This may pose a major design problem.

The pressure vessel top has a 4-ft diameter flanged opening. While no extensive study has been made of the control and fuel loading mechanisms, it appears to be feasible to perform the necessary operations through this size of opening.

The fuel is unloaded through a 16-in. diameter nozzle in the bottom of the vessel. This nozzle is directly connected to a spent fuel storage tank (Fig. 6). A valve is installed in the fuel discharge tube between the vessel and the fuel storage tank.

4. Shielding

The shielding for the high flux reactor has been estimated. These approximations lead to an over-all reactor diameter of 36 ft using heavy concrete shielding.

The conceptual shielding arrangement (Figs. 4-6) shows a void zone, approximately 1 ft thick, directly outside the reactor vessel. Whether the zone remains a void volume in the final design is contingent upon evaluation of the neutron density outside the reactor tank at times when various experiments are in the reactor. If a significant neutron flux level still exists under such conditions, then the zone or parts of the zone now shown as void may be utilized for irradiation facilities. If significant neutron fluxes do not exist outside the tank, or if experiments placed in these locations are likely to give interfering radiations to the major beam holes, the void zone will be used to attenuate the neutron flux by means of boral or boron-containing solutions.

Thermal shielding is located outside the neutron absorption zone. This may consist of a solid or split iron shield (Fig. 5) with a coolant provided in the annulus. A liquid coolant is preferred because the size of coolant supply lines required is much smaller than if a gas coolant is used.

The estimated thickness of concrete shield, which is of high-density concrete may not be the most desirable design thickness. A larger spacing, e.g., between beam holes at the outer periphery of the shield, could be obtained by use of lower density concrete, giving greater access to experiments.

5. Coolant Systems

a. D₂O Systems

The main D₂O system has a circulating rate of about 50,000 gpm, 45,000 gpm of which passes through the core; the balance bypasses through the moderator-reflector zones. The system consists of

two parallel interconnected flow circuits utilizing 24- and 30-in. pipe lines pumps of about 12 000-gpm capacity, and heat exchangers of 20,000 - 30,000 sq ft surface area.

As shown in the tentative flow diagram (Fig. 7), the primary D_2O flow enters through a 36-in. diameter nozzle which terminates in a flanged joint close to the bottom of the vessel. The nozzle is constructed of the same material as the vessel. Inside the vessel, the inlet line continues as a zirconium section which terminates at the lower fuel support grid. The flow continues upward through the core and exits through two 24-in. nozzles in the vessel wall near the top. External to the vessel, the flow continues through heat exchangers and then is pumped back into the reactor to complete the cycle. The inlet line forms a loop within the concrete shield. The loop is vented at the top to the reactor tank; thus a rupture in the D_2O lines, external to the shield, will not result in loss of water out of the reactor.

Provision will be made for a helium-circulating system to remove and disperse the radiolytic gases formed within the vessel. With reference to the system shown in Fig. 7, helium is circulated across the surface of the D_2O to remove the formed D_2 and O_2 . The gas stream then passes through a dryer to remove the entrained moisture, thence to a recombination catalyst chamber, and then to a cooler-condenser for removal of the recombined D_2O which then returns to the system.

The provision for gas removal is not necessarily limited to the confines of the reactor vessel. Further study may indicate that separators external to the reactor may be more advantageous. In this instance the D_2O lines would pass from the reactor into the separators wherein a helium stream would entrain the separated D_2 and O_2 into a dryer and then to a recombiner. The gas stream would be cooled and the D_2O condensate returned to the system. The bulk D_2O from the separators flows to the tubular heat exchangers where the heat is removed by the H_2O -circulating system. The D_2O is pumped from the heat exchangers back into the reactor.

Other D_2O -circulating streams are required in addition to the main coolant stream. The zone outside of the core requires a continuous supply of D_2O . This supply enters at the bottom of the vessel, either through a series of small nozzles around the bottom or through the side walls of the 36-in. inlet nozzle at a point directly inside the vessel bottom. This latter position is favored since the openings in the side walls of the main circulating lines are available for natural circulation flow through the core. Other provisions for D_2O flow include the side circulation loop which feeds from the side of the vessel near the top of the core and connects into the main inlet line. A check valve is located in this line. During normal operation the pump pressure holds the check valve closed. On loss of pump pressure, the check valve opens to form the main flow path for natural circulation.

Provision is made within the shield for emergency cooling when steam is generated in the reactor following loss of flow from pumps. The condenser is located in the circuit and the condensate returns into the bottom of the core (Fig. 8).

Auxiliary D_2O systems may be required for special cooling of experiments and possibly for some emergency uses. Cooling of the core during a power failure may be accomplished by either special pumps having a diesel-generator power supply, or emergency drive motors may be coupled onto the shaft of the regular circulating pumps. The latter would also be powered by diesel-generator sets. The regular circulating pumps have flywheels to maintain a reasonable flow rate during the period between the time of losing the normal power supply and the time in bringing the emergency, powered pumps up to speed.

It is recognized that some of the beam hole liners may require special cooling. No study has been made of the heat generation in the structural parts of the reactor.

Plans call for a D_2O -processing building (probably below ground level), divided into six, two-compartment cells. Each cell has one heat exchanger - pump combination. One of the compartments of each cell is shielded to accommodate a heat exchanger, a pump and required valving. The pump and valve drive mechanisms are located in the "cold" compartment of a cell. Four of the six cells are operating and two are standby.

The main D_2O -circulating system also contains apparatus to control water quality. The presence of the magnesium in the core requires that the D_2O be maintained at a pH of 9.5 - 10.0 in order to hold the magnesium corrosion rate at its lowest level. Magnesium hydroxide has sufficient solubility to maintain this pH. The system for control of D_2O quality is supplied with D_2O from the cold side of the heat exchanger. Normal pH control is by means of a by-pass flow through an anion resin bed. However, in order to control the concentration of the soluble cations, a mixed-bed resin column would take a smaller by-pass flow to completely deionize some of the D_2O . The deionized water has the pH adjusted with magnesium hydroxide before returning to the main D_2O -circulating stream.

The standard equipment, such as piping, pumps and instrumentation, should present no unique problems. It is quite probable that the designs and operating experience of the large D_2O systems at the Savannah River, NRX, and NRU reactors are directly applicable to this reactor.

b. H₂O System

No effort has been made to develop information or plans for the secondary coolant system. There are no features in the H₂O system that require other than standard commercial equipment. It is expected that properly treated water will be supplied to the tubular heat exchangers. The heat removed from the heat exchangers is then dissipated to the atmosphere through the use of cooling towers.

6. Fuel Handling and Storage

It is proposed that the actual loading and unloading machine will be an indexing mechanism which would replace the control rod top plug. This mechanism would be required to unlock and remove the upper core support grid, assuming that the latter is anchored to internal structural members. The unloading device would then be indexed to a fuel assembly, lift it clear of the core, rotate to a position above the fuel discharge tube, and then lower the fuel into a D₂O-filled fuel storage tank.

On reaching the D₂O-filled fuel storage tank, the fuel assembly is picked off of the unloading tool by an arm located on an endless conveyor chain. The chain is then moved forward one fuel assembly position in order to receive the next fuel assembly. The storage tank is designed to accommodate about three core loadings. The operation may be made fully automatic or, by use of remote viewing devices, may be a semi-automatic operation.

The tank for storage of D₂O and fuel is capable of withstanding the pressure caused by the head of D₂O in the reactor vessel. During unloading operations, the D₂O circulates from the reactor vessel through the discharge tube, into the storage tank, then to a pump, a heat exchanger, and back to the reactor vessel.

There will be forced circulation when fuel is in the storage tank. The storage period in this tank may be as long as 2.5 weeks. When it is desired to remove fuel from the D₂O tank the fuel assembly is moved to the unloading position which is either in a cubicle in the sub-reactor room shield wall or is directly adjacent to the wall. A shielded coffin is located over the unloading hole and a liquid-tight seal is made between the coffin and the fuel storage tank. The coffin is filled with D₂O and the assembly is grappled and drawn into the coffin. The coffin is disconnected from the D₂O tank and moved to a position where a connection is made with a storage tank filled with an organic liquid. The organic liquid is one which is immiscible with water, has an appreciable density difference from either D₂O or H₂O, and is not extremely sensitive to radiation damage. The organic liquid is pumped into the coffin displacing the D₂O. H₂O is then pumped in, displacing the organic liquid. This procedure should hold the isotopic dilution and physical loss of D₂O to a low level. A small clean-up system is

provided to purify D_2O which is contaminated with the organic liquid. The clean-up system is quite likely to use distillation and/or an absorptive bed of activated carbon or silica gel.

The coffin is moved to the fuel storage canal and the fuel assembly is dropped out of the bottom of the coffin into a depth of H_2O sufficient to shield personnel viewing the fuel from above.

The spent fuel may be used for irradiation facilities in the canal. A second interesting possibility exists for the utilization of the irradiated fuel. The fuel assemblies from this reactor form a high photo-neutron source when stored in D_2O . Fuel assemblies in their early stages of cooling can be placed in properly designed pots containing D_2O to form neutron test or irradiation facilities having neutron flux levels as high as $10^{10} \text{ n}/(\text{cm}^2)(\text{sec})$.

The fuel canal is sufficiently long to store all fuel which must be held in cooling storage, plus sufficient space for at least one major gamma-irradiation facility, and for the storage of miscellaneous irradiated materials.

7. Experimental Facilities

The experimental facilities have not been defined in number or in size. The guiding principle behind the design of the facilities is that they are to be primarily for research of a fundamental nature. The small, fully enriched core is designed to yield a high flux level. It cannot tolerate an appreciable flux perturbation in any major experimental facility without excessively disturbing the flux in adjacent experiments. All beam holes, through holes, and vertical access hole liners are of zirconium. This is the only structural material of low cross section which has a sufficiently low corrosion rate in alkaline water.

The facility with the highest flux is a vertical thimble located in the center hole in the core. This thimble is expected to have a maximum diameter of 2 in. The method of providing access to this thimble will necessitate extensive design study, especially since it is in a position where it poses difficulties in the design of the control and fuel-handling mechanisms. If the irradiations to be performed in the center thimble can be handled through a wide radius curve, then the central thimble may be designed as a semi-permanent structure with the liner being offset. The access hole on the top will be off to one side of the top plug. Should the offset of the thimble liner detract markedly from the utilization of the central thimble, then the thimble design involves a straight tube which may be incorporated in the top control plug. The latter arrangement may necessitate complete or partial removal of the central thimble during fuel unloading and loading operations.

11.

The horizontal beam and through holes (Fig. 5) are not formed by simple stepped tubular liners with simple shield plugs, but are quite complex facilities designed for rather specific types of experiments or apparatus. Vertical access holes, other than the center thimble, are located in the D₂O reflector, preferably away from the core where irradiations being performed in them have only small influence on horizontal beam holes. The vertical holes which terminate near the core are slanted to create straight hole liners which have the top access ports away from the top plug.

Figure 5 shows two graphite thermal columns. While thermal column demand and usage is high, the horizontal beam holes may be of more value in the high-flux reactor. While the final design may not contain two thermal columns, certainly the reactor must have at least one such facility. The requests for usage of this type reactor indicate that a larger thermal column than the ones shown may be desired. In addition, there is the possibility of using D₂O in the thermal column. Studies will be made to determine possible advantages and versatility of D₂O in lieu of graphite.

The manner in which the experimental facilities are used in the Mighty Mouse reactor requires an appreciable amount of survey and design study. It is necessary to develop a better understanding of the means of designing facilities for proper usage, the magnitude of the cooling requirements, the methods of supplying the coolant, and methods and weights of shielding required by experiments. This lack of development of the experimental facilities should in no way effect the feasibility of the reactor.

8. Reactor Building

The initial study of a reactor building involves a cylindrical steel shell, approximately 120 ft in diameter and 190 ft in over-all height. The distance from ground level to the lowest elevation of the shell is about 70 ft. The building should be capable of withstanding 3-5 psig internal pressure. The sidewalls need not be self-supporting but may be hung from the columns which also support the crane bridge. The main floor of the building and the first basement level are intended primarily for experiments. Any sub-basement levels will house spent fuel storage tanks, canals, pipe lines containing active fluids and operating equipment.

Control and instrument rooms may be carried off the side walls of the steel shell either internally or externally. If the control room is external to the side wall, then viewing ports or windows will be installed so that the reactor and most of the main floor are visible from the control room. Tentative openings in the shell include at least one vehicle door and two personnel-access doors.

The storage canal for spent fuel extends beyond the confines of the cylindrical wall of the building. The canal and the canal housing would be constructed to withstand the same internal pressure as the main building. Any experiments which require greater linear distance away from a reactor face than can be had inside the building can be housed in tubular extensions added to the building directly in line with the particular beam hole to be used and running radially outward from the building.

B. Physics

The demand for high flux per unit power leads to consideration of D_2O -moderated thermal reactors, since D_2O supports a higher thermal flux per unit power than any other moderator in use. There are alternate systems - fast, mixed fast-thermal, homogeneous circulating fuel, or reflector moderated - which offer the advantage of improved heat-transfer characteristics, thereby allowing less leaky configurations. In the case of the fast and the mixed fast-thermal systems, the ratio of peak thermal flux to power is increased. From the standpoints of engineering and physics, however, considerably less is known about these alternate systems, so that the more conservative approach is to consider a D_2O system similar to CP-5.

The physics calculations needed for a preliminary survey are parametric surveys which will answer the following:

1. What is the core configuration which has the maximum value of peak flux per unit power?
2. What type of perturbation is caused by experimental apparatus, control rods, and structural elements, and what amount of control is necessary?
3. Is there a possibility of an auto-catalytic reaction wherein a power rise would cause xenon-135 to burn up, thereby increasing the reactivity and, consequently, raising the power level?

The first two questions can be answered approximately by a series of critical calculations, and these are outlined in the following section. The third question, fortunately, is qualitative in nature and can be answered by calculating the xenon and iodine concentrations in the reactor under a variety of operating conditions.

1. Critical Studies

The critical calculations were of the two-group variety, since it was felt that this approach, coupled with CP-5 experience, would

yield adequate answers for a preliminary survey. Heat transfer characteristics dictate the core volume to a large extent. The annular shape of the core was suggested by the fact that the increase in critical mass over a cylinder is more than compensated by the peaking of the thermal flux in the internal reflector. This peak does not change much once the internal radius is of the order $2\sqrt{\tau}$, where τ is the slowing down area. In all calculations the internal core diameter was assumed to be 30 cm. The internal radius is slightly less than $2\sqrt{\tau}$, but since the core is dilute there is some moderation in the core itself. The height and thickness of the core were varied, the height varying between 75 cm and 90 cm, and the thickness between 22.5 and 35 cm. The structural material envisaged was aluminum or magnesium, and the moderator pure D_2O . The fuel was assumed to be pure U^{235} . With the magnesium structure, it was assumed that 1 part per hundred thousand atoms of the uranium was xenon-135, since this is the approximate amount of equilibrium xenon in such systems. The capture cross section of aluminum is great enough to make the xenon poisoning a smaller effect. Table II lists the core parameters and their ranges.

Table II

CORE PARAMETERS AND THEIR RANGES

Parameter	Range
Core Shape	Homogeneous sphere - cylindrical annulus
Metal/ D_2O (by volume)	0 - 0.25
Reflector Thickness, cm	75 - 115
Core Thermal Capture Cross Section, cm^{-1}	0.004 - 0.015
Outer Core Radius, cm	37.5 - 50
Core Height, cm	75 - 90
Structural Material	Al - Mg*
Burnup (at constant power of 250 mw), gm	0 - 1550

*Dow Chemical Co. A.E.C. Alloy

Two-group constants were calculated in a conventional fashion except for the age in the reflector. Experience with matching D_2O two-group parameters with CP-5 data led to the suggestion that the two-group age in D_2O in the reflector be taken to be 200 cm^2 . The Fermi age, τ_f , of a metal- D_2O mixture was calculated numerically. A useful

approximation for dilute mixtures is to compute the age as a universal function of D_2O density and excess scattering per D_2O molecule; this was done graphically and the results are in close agreement with the more accurate numerical integration. The fast diffusion coefficient was calculated similarly. Cross sections as a function of energy were taken from BNL-325.³

Thermal cross sections were adjusted for a Maxwellian neutron velocity distribution with a neutron temperature corresponding to 0.025 ev. The thermal diffusion length in D_2O was assumed to be 115 cm. This corresponds to slightly impure D_2O . D_s was calculated from values of σ_s reported in BNL-325. For a system of material buckling, B^2 , the two-group age, τ_{2g} , was found by solving:

$$e^{\tau_f B^2} = 1 + \tau_{2g} B^2 .$$

Admittedly, there are deficiencies in this mode of calculation, but previous experience indicates that two-group calculations are sufficient for the present purposes. Table III is a compilation of two-group constants used in the calculations reported here.

The first problem solved was that of determining approximate values of the reflector savings as a function of reflector thickness and core composition. Since the proposed core is a right cylindrical annulus, it appears that the solution of an appropriate spherical problem would yield reasonable estimates of the reflector savings.⁴ Subsequent calculations revealed that these estimates are very good. The problem solved was that of finding the critical composition of a sphere of D_2O and aluminum in a fixed ratio together with U^{235} in an unknown amount and the spheres reflected uniformly by D_2O . The sphere was assumed to be homogeneous and included the entire volume within the outer surface of a proposed core, i.e., the inner reflector was homogenized with the sphere. This was done for a sphere of 43.6 cm radius. No xenon-135 or other poison was included in these calculations. The results are plotted in Figs. 9 to 11. As can be seen from Fig. 10, the reflector savings are not greatly affected by the thermal capture cross section of the core, so that the omission of xenon or other poisons does not lead to large errors. These results substantiate the concept of using a very thick reflector, since the reflector savings then amount to about 44 cm. A stronger argument for the proposed reflector thickness (150 cm) is that the flux out to this point is of the order

³D. S. Hughes and J. A. Harvey, "Nuclear Cross Sections" (1955).

⁴Reflector savings values are to be added to the physical diameter of the core, i.e., they are twice the usual amount.

of the peak MTR flux, so that the outer reflector provides a large volume in which neutron fluxes ranging upward from MTR peak fluxes are available for experiments. The cadmium ratio at the outer boundary is of the order of 10^4 and this is of some advantage both for shielding and for thermal column work.

Table III
TWO-GROUP CONSTANTS

Material	Fast			Slow				η
	D_f , cm	$\tau_f(\text{Core})$, cm ²	$\tau_{2g}(\text{Reflector})$, cm ²	D_s , cm	L , cm	Σ_a , cm ⁻¹	Σ_{tr} , cm ⁻¹	
D ₂ O	1.25	130	200	0.75408	115			
Al						0.01229	0.0823	
Mg						0.00292	0.16704	
U ²³⁵ *						29.52708		1.98866
Al:D ₂ O = 0.1	1.309	150						
0.2	1.380	170						
Mg:D ₂ O = 0.1	1.287	147						
0.2	1.320	164						

*Includes 1 atom Xe¹³⁵/10⁵

Solution of the critical problem for an annular core is slightly more difficult than for the simpler geometries, and a numerical method susceptible to use with a desk computer was derived. The method (see Appendix B) is similar to numerical methods used previously except, perhaps, for its treatment of boundary conditions. It is at present being adopted for use on large computing machines. Use of this method with aluminum as the structural material yielded an estimated critical mass of 2.2 kg for the reactor in a cold, clean condition.

Since the high burnup implies a short cycle time, magnesium was substituted for aluminum in an effort to reduce the critical mass and thereby increase the peak flux per unit power. Heat transfer and design considerations indicated that a Mg:D₂O volume ratio of 0.24452 would be within a satisfactory range. The core is finely divided and assumed to be homogeneous. A survey was made of the flux per unit power as a function of outer radius; the inner radius was fixed at 15 cm, the height at 90 cm. The results of the survey are shown in Fig. 12. There is a small, but definite, maximum at an outer radius of 45 cm. The characteristics of

the optimum core are listed in Table IV, together with the effects of various perturbations. The latter were calculated by conventional perturbation theory. Flux plots not including the photo-neutron contribution are shown in Fig. 13.

Table IV

CRITICAL PROPERTIES OF OPTIMUM Mg CORE

Condition	Remarks
Loading after 120 hr operation; no fission products, 1.46×10^{-5} atom Xe/atom of U^{235}	1344 gm U^{235} needed for criticality.
Burnup for 120 hr operation at 250 mw	1550 gm U^{235}
U^{235} needed to compensate for non-saturating fission products after 120 hr operation (on the basis of 50 b/fission)	341 gm
Burnable poison necessary to compensate for burnup and fission product allowance, but not xenon	2163 cm ²
Uniform addition of mass to burned-up core	$\delta\nu/\nu = 0.21099 \delta m/m$
Uniform void in burned-up core	$\delta\nu/\nu = -0.46 \times 10^{-6}/\text{cc void}$
Addition of thermal poison to burned-up core	$\delta\nu/\nu = -0.4004 \times 10^{-3}/\text{cm}^2$ of absorber
Addition of thermal poison to fresh core	$\delta\nu/\nu = -0.19648 \times 10^{-3}/\text{cm}^2$ of absorber
Peak thermal flux to average thermal flux in burned up core	1.79222

2. Xenon Problem

Appendix A summarizes the preliminary evaluations of xenon and iodine concentrations. These concentrations were found for three different reactors so that sound results might be obtained at a time when the reactor composition was extremely tentative. Of particular interest are the calculations of k_{eff} under conditions of a large instantaneous increase in flux level. These calculations showed that the increase in k_{eff} attendant upon an instantaneous power increase is at the rate of about 0.4% per hr, and such a rate is certainly controllable by the reactor controls. A brief examination of the problems arising from power oscillations indicated that no control problems lie in this direction either. Finally, a comparison of operation at constant flux and constant power showed that the latter condition is slightly easier to control, especially in the case of power sags, but the advantage is not great. Operation at constant power was assumed in subsequent calculations.

3. Control

As can be seen from Table IV, control by voids or conventional neutron absorbers is feasible; however, absorbers waste neutrons which could be used more advantageously for experiments; concomitantly, they burn out rapidly. Control by addition of burnable poison to the fuel is discussed in Appendix C. The introduction of experimental apparatus affects reactivity, usually adversely. With reference to Table IV it can be determined that 833 gm of stainless steel are worth about 1% in reactivity; thus the apparatus must be carefully designed to use a minimum of absorbing materials.

The temperature effect has not been calculated, but should be similar to that in CP-5.

The operating cycle time of 120 hr, while arbitrary, is based on the following observations:

- (1) Fission product containment should not pose an unusually large hazard in the event of an excursion. After 120 hr the 1550 gm of fission products are within an order of magnitude of those contained in CP-5.
- (2) Five days is a convenient operating cycle.
- (3) Additional fuel becomes increasingly difficult to control; and there is a corresponding decrease in flux.
- (4) Fission product poisoning with longer cycle times represents an increasingly large amount of parasitic capture.

4. Shielding

Shielding difficulties are mitigated, to some extent, by the large tank of D_2O which attenuates the fast neutron flux very rapidly.

The thermal flux penetrating the reactor vessel wall can be absorbed in boron or in materials used for the production of isotopes.

From experience with CP-5, it is expected that after shut-down for two days, the core should produce a photo-neutron flux of about $5 \times 10^9 \text{ n}/(\text{cm}^2)(\text{sec})$. While not considered a problem in shielding the reactor vessel, the photo-neutron flux creates a problem with respect to the D_2O piping and the D_2O -filled fuel storage tank. Therefore, the shielding will be similar to that of CP-5 or, at most, of the MTR. No detailed calculations have been made.

C. Heat Generation and Removal

1. Steady-state Conditions

The energy input to the coolant at full power and at steady-state conditions is 250 mw. Severe problems of fuel design and heat are imposed by the requirements of removing this quantity of energy as heat from the small core volume proposed for Mighty Mouse:

- (1) Low operating pressures are required to avoid heavy pressure vessel and flange construction, as well as excessive gamma heating of test thimbles and flux depression due to the presence of large amounts of absorber.
- (2) A low temperature of the coolant-moderator and of the fuel elements is desired to provide additional thermal capacity during power transients and to reduce structural design problems.
- (3) A secondary system, circulating light water and dissipating heat in a cooling tower, must be matched with the primary system. Considerations of cooling tower size force inlet temperatures upward to reduce the size or number of cooling towers.

The results of such considerations, without detailed optimization, are shown in the tentative system flow diagram, Fig. 7. The pertinent primary coolant system data are summarized in Table V.

Table V

SUMMARY OF CONCEPTUAL CORE DESIGNOver-all Performance

Thermal Power, mw	250
Power Density, kw/liter	498
Core Life, hr	120

Mechanical Characteristics

Geometry. Right circular annulus	
Active Diameter, cm	
Inside	30
Outside	90
Active Height, cm	90
Reflector. D ₂ O annulus, 90 cm ID x 13 ft OD	
Fuel Elements	
Geometry: Curved plates; axial flow	
Plate Thickness, in.	0.030
Core Composition	U-Mg dispersion
Clad, in.	0.010
Clad Material	Mg

Heat Transfer Characteristics

Nominal operating pressure, psia	60
Saturation temperature, F	293
Coolant inlet temperature, F	140
Coolant outlet temperature (avg. channel), F	175
Peak coolant temperature, F	183.8
Average coolant temperature, F	157.5
Max. surface temperature (hot channel), F	269
Max. center temperature (hot channel), F	~ 276
Heat transfer area, sq ft	1800
Flow area, sq ft	4.48
Average coolant flow velocity, fps	21.4
Coolant flow rate, lb/hr	24.38×10^6
Pressure drop, psi	1.75
Heat Flux, Btu/(hr)(sq ft)	
Core Average	475,000
Max. at minimum burnout point	633,000

Table V (cont'd.)

Heat transfer coefficient, Btu/(hr)(sq ft)(F)	6,880
Max./Avg. Flux Distribution	
Radial	1.19
Axial	1.11
Hot Spot factors*	1.25
Coolant temperature rise	1.25
Film temperature rise	1.19
Heat flux	1.13

*Estimated from data of a previous reactor study. Differences between the two cores adjudge the estimates as conservative.

a. Thermal Analysis of Proposed Core Designs

Heat may be successfully removed from the core by a number of configurations, all of which approximate the homogeneous fuel and moderator distribution assumed for physics calculations.

In the axial flow, tubular element core (Fig. 1), the coolant flows both inside and outside of the tubes. This arrangement offers a considerable amount of heat transfer area and an advantage in hot spot and hot channel characteristics. The advantage is that the coolant flow outside the tubes may mix freely; thus, cooling of the outside surface of the tubes is not hindered by momentary stoppage or restriction of flow inside the tubes. Restriction of flow outside the tubes limits rising temperatures to the local zone, beyond which cross mixing of flows and cooling continues in the normal manner. There are parallel channel effects to be considered; however, the severity of any situation arising from this source will be less than in channels without mixing of flows.

The axial curved plate core (Fig. 3) is similar to the MTR-type fuel element, except that the manner in which the plates are stacked in the fuel boxes precludes vertical alignment of the respective channels to form continuous passages for coolant flow. The manner in which openings between channels occurs does not aid in raising the convective heat transfer coefficient. Some communication is provided between adjacent rows of plates for flow out of, or into, momentarily blocked channels, which helps to prevent channel burnouts. Also, the possibility exists that the advent of localized, permanent channel obstructions may effect lower plate temperatures and, hence, retard burnout.

b. Calculations

(1) Distribution of Energy Input and Temperatures

The flux distribution for thermal neutrons determined from two-group calculations is shown for the radial direction in Fig. 14. The axial distribution is approximated by a cosine distribution with 44-cm reflector savings. The energy spectrum is such that fast fission is neglected for thermal design of the core.

The fluid temperature distribution for axial channels with no mixing between channels is given by the relation:

$$T_x = T_i + \frac{L_{rs} Q (\max)}{\pi W c_p} \left[\sin \left[\frac{\pi(X - L/2)}{L_{rs}} \right] + \sin \left(\frac{\pi}{2} \frac{L}{L_{rs}} \right) \right]$$

where

T = Temperature, F

L_{rs} = Active length of core plus total reflector savings, ft

Q = Heat generation per unit length of channel, Btu/(hr)(ft)

W = Coolant flow, lbs/hr

c_p = Specific heat at constant pressure, Btu/(lb)(F)

X = Distance within medium under study, ft

L = Active length of core.

The subscript x refers to a local point, and the subscript i to an initial point.

The surface temperatures of these channels are given by:

$$T_S = T_x + \frac{Q'' (\max)}{h} \cos \left[\frac{\pi(X - L/2)}{L_{rs}} \right]$$

where

Q'' = Heat flux, Btu/(hr)(ft²)

h = Convective coefficient of heat transfer, Btu/(hr)(sq ft)(F).

The subscript S refers to the fuel element surface.

The maximum temperature of the metal is obtained from the relation

$$-k \frac{d^2 T}{dX^2} = Q \quad ; \quad 0 \leq X \leq a \quad ;$$

where

k = Thermal conductivity, Btu/(hr)(ft)(F)

a = Half thickness of section containing fuel, ft.

The above relationship holds for the fuel-bearing portion of an infinite slab with uniform heat generation, Q . Conduction through the cladding follows the relation

$$Q''' V_f = -k A \frac{dT}{dX} \quad ; \quad a \leq X \leq b \quad ,$$

where

Q''' = Volumetric rate of heat generation, Btu/(hr)(cu ft)

V_f = Volume of fuel/element, cu ft

A = Heat transfer area, sq ft

b = Half thickness of the fuel plus thickness of the clad, ft.

Solution of the first differential equation for $\frac{dT}{dX} = 0$ at $X = 0$ and evaluation of the second gives

$$T_0 - T_s = \frac{Q''' a^2}{2k} + \frac{Q''' V_f}{A} \left(\frac{b - a}{k_c} \right)$$

where the subscript 0 refers to the fuel center line, and the subscript c to the clad.

Since

$$Q_x = Q_x (\max) \cos \frac{\pi[X - (L/2)]}{L_0}$$

and

$$Q_x L / V_f = Q,$$

then

$$T_0 = T_i + \frac{Q_x(\max)L_{rs}}{\pi W c_p} \left\{ \sin \frac{\pi[X - (L/2)]}{L_{rs}} + \sin \left(\frac{\pi L}{2L_{rs}} \right) \right\} \\ + Q_x(\max) \cos \frac{\pi[X - (L/2)]}{L_{rs}} \left[\frac{L}{V_f} \frac{a^2}{2k} + \frac{2}{A} \frac{(b-a)}{k} + \frac{L}{Ah} \right]$$

and the point of maximum temperature

$$x = \frac{L_{rs}}{\pi} \tan^{-1} \left[\frac{L_0}{\pi W c_p \frac{L}{V_f} \frac{a^2}{2k} + \frac{L}{A} \frac{(b-a)}{k} + \frac{L}{Ah}} + \frac{\pi}{2} \frac{L}{L_{rs}} \right]$$

A representative set of calculations for a curved plate core design with axial flow has been made. The data are given in Table V. The power and temperature distribution are plotted in Figs. 15 and 16, respectively. Convective heat transfer coefficients for parallel channels are given as:

$$h = 0.023 (k/D) (N_{Re})^{0.8} (N_{Pr})^{0.4} ,$$

where

D = equivalent diameter, ft

N_{Re} = Reynolds number

N_{Pr} = Prandtl number.

Figure 16 shows the surface temperatures as a function of channel length for full power operation. The maximum temperature distribution is shown in the uppermost curve. Under these conditions there is no local boiling in the core.

A similar temperature distribution is expected for the tubular type of core; however, some differences will arise due to distribution of flows between the shell side of the core and within the tube.

(2) Burnout

Burnout data are not available for the specific situations considered in this report. However, the order of magnitude of the burnout heat flux can be determined from the data of McAdams⁵ and

⁵W. H. McAdams, et al, "Heat Transfer at High Rates to Water with Surface Boiling," Ind. Eng. Chem., 41, 1945-1953 (1949).

others.⁶ Table VI lists burnout heat fluxes, indicated by McAdams, for channels of 0.52-in. equivalent diameter and various subcoolings at 60 psi. The equivalent diameter of the curved plate core (Fig. 3), is 0.36 in.

Table VI

CHANNEL BURNOUT DATA

$D_e = 0.52$ in.

Pressure = 60 psi

Velocity, fps	Subcooling, F	Burnout Heat Flux Btu/(hr)(sq ft) $\times 10^{-6}$
12	100	2.0
	50	1.4
4	100	1.3
	50	1.1
1	100	0.85
	50	0.7

The data in Table VI pertain to an annuli of relatively low values of L/D . Leyse⁷ has deduced an L/D effect based on a number of experiments. His empirical relation is:

$$Q''_{\text{burnout}} = 2.5 \times 10^5 \left(\frac{V \Delta t_{\text{sub}}}{L/D} \right)^{0.43},$$

where

V = velocity, fps

L = channel length, ft

Δt_{sub} = subcooling, F

This relationship has been used to determine the minimum allowable burnout heat flux. Application of the L/D correction factor to the data in

⁶F. Kreith, M. Summerfield, "Heat Transfer to Water at High Flux Densities With and Without Surface Boiling," Trans. ASME, 71, 805-816 (1949).

⁷C. F. Leyse, "Permissible MTR Power Levels With and Without Film Boiling," IDO-16064 (January 15, 1953).

Table VI indicates a safety factor of two in the burnout point under the proposed reactor operating conditions. Increase in velocity and/or system pressure will increase the factor.

The calculated maximum heat flux in the core is 0.63×10^6 Btu/(hr)(sq ft). The subcooling at this minimum burnout point is slightly greater than 100F.

c. Associated Heat Removal Problems

A central experimental thimble, when inserted, will create a heat source which must be considered. Cooling of this thimble is expected to be accomplished by by-passing a portion of the main coolant flow along the thimble in parallel with the main flow circuit. Additional cooling, for cases where heat generation is higher than the capability of such a system, may be accomplished by an external cooling system contained in the thimble. A limited amount of cooling of the pressure vessel and the thimbles inserted through the reflector may be accomplished with the circulation of some of the reflector D₂O. External cooling will have to be used for experiments having large heat generation.

Because of the high neutron flux at the wall of the pressure vessel, the heating of the thermal shield by neutron and gamma absorption is expected to be quite large. Probably the most advisable type of cooling for this region of the structure will be by the passage of water over well-dispersed cooling surfaces. The volume of liquid cooling ducts will be small; hence no interference with experimental facilities is anticipated. Cooling of the biological shielding concrete may be accomplished by cooling water pipes imbedded in the shield concrete at its inner surface.

2. Unsteady-state Conditions

The cooling system for the reactor plant is provided with a number of safeguards to forestall damage to the reactor during unsteady-state situations occasioned by.

- a. normal reactor shutdown;
- b. loss of power, or pump failure in either the primary or secondary system; or
- c. rupture in primary coolant system.

a. Normal Reactor Shutdown

When the reactor is shut down after operating at full power for a period of 5 days, a considerable quantity of decay heat must be dissipated. For example, at the end of one day after shutdown, the decay heating still amounts to 0.5 mw (see Fig. 17).

Normal shutdown allows the coolant flow to be reduced by use of an auxiliary diesel-generator-powered pump which is provided to insure adequate circulation until natural circulation through the reactor tank will cool the core. The natural circulation loop, shown adjacent to the pressure vessel (Fig. 8), is so arranged that a check valve opens when pump circulating head is removed, and a flow is established by virtue of the density difference between the liquid in the core and that in the external circulating leg. The time delay between reactor shutdown and removal of the core will depend on the ability of the spent fuel-removal system to remove the decay heat. Scrams which originate from operational or experimental sources may be arranged so that coolant circuit flow is unchanged and hence may be handled, insofar as the coolant system is concerned, as normal shutdowns.

b. Loss of Secondary or Primary System Flow

Circulating pumps may be arranged with an auxiliary pumping source, e.g., diesel engines which float on the line under normal operating conditions. Failure of both these systems initiates flow from the emergency cooling supply tank to the emergency cooler and then to the heat exchanger.

Primary coolant pumps are arranged, as in the case of the secondary system, to have an auxiliary power supply. In addition, a flywheel of moment of inertia sufficient to store energy that will maintain cooling water circulation during the scram period and the initial part of the decay period is required. When the flywheel speed is reduced to the point where it no longer supplies pressure head for circulation of the fluid, the check valve on the return lines from the emergency cooling tank to the core inlet will open. The heating of the liquid, perhaps with the production of vapor if power density and flow prescribe it, will then establish natural circulation of fluid through the core to the emergency cooling tank. As long as the secondary cooling system is maintained in operation, cooling water is supplied to the emergency cooler. If the secondary system supply is lost as well, flow from the emergency cooling tank is initiated.

c. Rupture in Primary Coolant System

Rupture of the reactor pressure vessel or of any piping downstream of the inlet standing leg must be considered catastrophic. However, rupture of the primary system upstream of this point will allow natural circulation of primary coolant through the emergency cooler. The energy input to the cooler is dissipated through the secondary cooling system.

III. ALTERNATE DESIGNS

A. Core

An alternate core design under consideration consists of flat plates in radial array, supported and spaced in magnesium combs (Fig. 18).

The coolant flows axially into the top and/or bottom of the center annulus of the core (thus cooling the central thimble at the same time), and then radially outward, using the reflector section of the reactor vessel as an exhaust plenum. Flow mixing occurs in annular spaces between the respective rings of fuel elements.

The hot channel factors for such a core are considerably less important than for the continuous flow core. An advantage is also obtained in the burnout heat flux level in that the values of L/D of the channels are much less than for an axial flow core with the same plate spacing. While the radial flow produces an improvement in certain flow and heat transfer characteristics, the center opening in the proposed core is too small to supply a suitable quantity of D_2O at reasonable flow velocity within the core center. Should additional studies result in doubling the cross-sectional area of the central core opening, the radial coolant flow pattern may be given further study. However, this flow has one basic draw back; emergency cooling by natural circulation is more difficult than with an axial flow core.

B. Reactor Vessel

The problem of designing, fabricating, and installing a good thermal column window may necessitate further study of an alternate design for the pressure vessel.

The alternate design proposes to use a zirconium vessel having an inside diameter only as large as necessary to accommodate the core and essential control or fuel-handling mechanisms. In effect, the core would be a restriction in the main coolant line.

The core vessel, in turn, would be surrounded, radially, by an annular, aluminum vessel containing the D_2O reflector and accommodations for experimental facilities, such as beam hole liners and thermal columns.

Each vessel would be serviced by a separate D_2O supply and control system for water quality. The operating pressure of the reflector D_2O would be very low; hence the wall of the aluminum vessel would be thin. No special thermal column window would be required since aluminum or zirconium are acceptable materials.

While the two-vessel proposal appears to offer some advantages of easing design and materials problems, one of the disadvantages has served to limit study of the proposal. In the single vessel proposal, a large mass of D_2O is always available for cooling the core by natural circulation during emergencies arising from accidents outside the shield. This desirable feature could be achieved in the two-vessel system by cross-connecting, within the reactor shield, the two D_2O systems, with the proper valves and instrumentation necessary to guarantee low interleakage during normal operation. Such a feature represents a design problem and increases the response time for initiating natural circulation over that found in the single-vessel system. Intermixing the D_2O of the two systems could cause corrosion damage to the core due to the difference in water quality between the two circulating systems.

IV. RESEARCH AND DEVELOPMENT PROGRAM

A. General

Various phases of research and development are self-evident, although the order in which they are to be carried out has not been determined. In most cases, they may be carried out concurrently.

Under the general program there are two distinct survey tasks. The first is the establishment of the conditions imposed by various experiments and the development from these conditions of reference design specifications. The survey may be rather involved, since the experimenter frequently knows only the benefits which may accrue from the proper utilization of the high flux. How utilization is to be made of a particular experimental facility may have to be developed for the experimenter.

The second survey involves an evaluation of available technology and operating experience with research reactors. For example, this program can benefit from the operations of the MTR in terms of, say, their experience in handling experiments in and out of a high-flux reactor, the amount of building space, service facilities, and personnel required for major experiments.

The experience of Canadian research programs with the NRX reactor and their design philosophy in building the NRU reactor may be the most beneficial. Not only has a single group designed and built two major reactors, but both involve the use of D_2O .

These general survey studies should be made as the first stage in the development program.

In an effort to define the magnitude of the Mighty Mouse project in terms of U^{235} , a crude appraisal was made of the possible quantities consumed and tied up in process inventories. The following assumptions were used:

- (1) The fuel required per weekly cycle is 3.6 kg of 90% U^{235} .
- (2) The scrap and rejects in all processes from production of 90% U^{235} to the final fuel assemblies is such that the net yield is 50%. Scrap recovery takes 10 weeks.
- (3) Ten weeks inventory of fuel assemblies is provided.
- (4) 2.2 kg of U is recovered from spent fuel. Recovered U contains 70% U^{235} , the remainder being U^{236} and U^{238} . Spent fuel is 26 weeks in cooling and process.

Using these assumptions, an inventory of 156 kg of 90% U^{235} is built up in a period of approximately 40 weeks before an equilibrium situation is established. At equilibrium 3.6 kg of 90% U^{235} is removed from stock each week and 2.2 kg of 70% U^{235} is returned; in terms of annual quantities, these are 187 and 114 kg, respectively.

A knowledge of the magnitude of the quantities involved should aid in establishing whether new facilities are required for the various steps in the fuel cycle and, if new fuel facilities are required, where they are to be located. As fuel quantities are more firmly established, the quantities of fuel required for the cycle will be made more firm.

Factors other than fuel also influence the choice of site and of facilities for a site. Some of the factors are availability of water, electricity, manpower, the meteorology of the area, etc. Many of these factors have been examined, briefly, with respect to Argonne. However, the recognition exists that a more thorough survey and study is a necessary part of the Mighty Mouse development program.

B. Physics

Future physics investigations will include more accurate critical calculations, probably of the multi-group variety and perhaps two-dimensional. An attempt will be made to calculate the effects of burnup more precisely; the flatness of the flux across the core should facilitate this. The approximate calculations for the burnable poisons necessary will be refined and extended. As engineering design progresses, attempts will be made to assess the effects of structural materials and experimental apparatus. Shielding calculations must be included and these will involve extensive numerical calculations.

The theoretical studies can be complemented by an experimental program. Without referring to such problems as fuel fabrication, this program will include multiplication experiments to determine critical loadings, the worth of regulating and control rods, and critical experiments to determine operating characteristics, preferably through simulated operation over a complete core cycle. Such investigations can be carried out in the equipment envisaged in the Thorium-Uranium-Deuterium (THUD) program and in the 10-ft tank presently used with ZPR-V. Particular attention must be paid to the effects of experimental and structural apparatus on neutron flux distribution. The high flux desired necessitates the use of D_2O . Inadequate attention to the importance of absorbing materials in the moderator and reflector regions can negate the effort that has gone into producing the high flux.

Finally, paper studies of alternate systems will be carried out.

C. Heat Transfer

The problems which should be investigated are listed in order of their preference and value to the over-all program:

- (1) The heat transfer and fluid flow characteristics of the final configuration of the core should be studied experimentally. This includes a study of the mixing between channels where this is possible in the element design.
- (2) A study of local boiling and burnout characteristics of the fuel elements in water of the quality to be used in the reactor.
- (3) The effect of parallel channel flow instability upon operation of the particular core configurations.
- (4) The large amounts of energy to be dissipated after normal reactor shutdown and during emergency cooling operation.
- (5) A thorough investigation must be made of the unusually severe heating of the neutron and gamma-shielding accompanying the high flux of this reactor.
- (6) An extensive investigation of the thermal stress problems arising from gamma heating of the thimbles and experimental facilities.

D. D₂O Quality Control and Corrosion Studies

Control of water quality has been studied quite thoroughly for reactor systems having aluminum, zirconium and many ferrous alloys. In the aqueous, heterogeneous reactors built, the water quality has been kept high and either at or near the neutral point for reasons of maintaining low corrosion rate or low water activity resulting from dissolved or suspended materials. Since corrosion data on existing magnesium alloys shows the lowest corrosion rate in the pH range 9-10, some development work is necessary on control of water quality in this range.

This program should not require the development of new materials; rather, it involves the test of existing ion exchange resins and the combination of resins to determine the most suitable materials and operating procedures for maintaining proper water quality. It is known that commercially available anion resins can be used to maintain some degree of water quality control in the desired pH range. However, the use of only this type of resin will probably give insufficient control of the cation content

of the water. Tests are required to determine the degree of quality control with anion resins alone. Should this be insufficient, as is expected, then the test program must be extended to include the methods of control of types and concentrations of the soluble cations.

The filtering action of the ion exchange resins may be all the filtering required for the D_2O stream. Proof or disproof of this is another facet of the water quality test program. Should additional filtering be required, commercially developed filters should be adequate.

More test work is required to determine the corrosion characteristics of magnesium. Some fundamental work is needed to establish both the types of corrosion attack and the optimum aqueous environment for minimum corrosion rate. In addition, dynamic corrosion tests must be conducted to establish the effects of velocity, temperature, pH on magnesium and couples of magnesium with dissimilar metals.

In addition to studies of magnesium corrosion, tests of zirconium and zirconium-ferrous couples are required. In the critical zones of the reactor it appears that zirconium may be the only suitable low-cross section, permanent structural metal.

The corrosion rates of ferrous alloys other than the Type 300 series stainless steels should be investigated. In an aqueous medium as alkaline as is expected, many lower cost, more easily workable ferrous alloys may have acceptably low corrosion rates.

E. Fuel Fabrication and Processing

Of the two problems, fabrication and processing, the former appears to be the major one. Fuels containing highly enriched uranium in conjunction with magnesium have not been developed. Thus, the first stage of fuel fabrication program is the development of the technique for manufacture of fuel elements. The proposed short life of the core suggests that the fuel be assembled into large pieces to minimize fuel unloading and loading operations. The departure from more conventional square or round fuel assemblies may require extensive research and development work.

Uranium does not alloy with magnesium; consequently, the uranium must be added as a layer within the magnesium cladding or be dispersed throughout a magnesium matrix. The uranium may be in any of a number of forms, the more common ones being metal, oxide, carbide, or silicide. Uranium as metal might be deposited on a surface by vapor deposition; in any form it might be dispersed in a fluid and painted on a magnesium surface. Following the surface application of the uranium, a rolled-on cladding of magnesium may be applied. The uranium might be dispersed in some form in magnesium powder and this, in turn, canned in magnesium to make a billet

for extruding plates or tubes. The quantity of uranium necessary in this fuel is quite small; for example: if fully enriched uranium in metallic form were used in fuels shown in Figs. 1 to 3, a layer approximately 0.1 mil thick is required. It may well be that as a part of a fuel element program, some work may be necessary on developing suitable magnesium alloys. At the most, this is probably not a major program for the developments in the United States, and those in the United Kingdom are advanced.

The development work required for the fuel assembly may include:

- (1) development of casting alloys and techniques to obtain satisfactory precision castings for support and spacer grids;
- (2) development of machines and/or techniques for welding or fastening the parts into assemblies. Much of this work can be developed by private industry and may require only conventional operating practice using more precise controls.

In the development of both the fuel element and the fuel assembly techniques, the current technology of magnesium and the choice of avenues of approach for solving remaining problems is sufficiently great that no extensive development program is envisaged.

The processing of irradiated fuels for the purpose of recovering the residual, highly enriched uranium from the fuels containing a fairly large bulk of magnesium is not expected to be a major departure from the processing used for the aluminum MTR-type fuels. The dissolution of the magnesium fuels should be no problem. The separations chemistry of the resulting magnesium-containing solution is expected to be quite conventional. However, tests must be carried out to prove or disprove this. Assuming that no major differences are found, the remaining problem is to ascertain whether these magnesium fuels can be fed directly into existing or planned process plants, or what head-end treatments are necessary to make feed solution for existing or planned process plants, or whether a new process plant is required for magnesium-containing fuels.

F. Design

Quite naturally at this stage of development the entire reactor system is a development program. The reactor system has a solid background foundation on which to build. Operating reactor systems such as the MTR, NRX and CP-5 form this background. However, many characteristics change in going to high flux and high power. In addition, the demands of research personnel also change.

The design problems to be solved cover the fuel, the fuel-handling system, the control systems, the internal reactor vessel structure, the reactor vessel, the experimental facilities, the various shields, the process or cooling systems, the main reactor building with its various service facilities and, finally, general site development. For many of these items the development work is of a conventional nature. For example, the reactor vessel and the shields, while they are probably complex, should not require a development necessitating mock-ups, new materials or new construction techniques. The cooling systems can follow the practice used for the design of D₂O reactors such as CP-5, NRX, or NRU. The buildings and site development should follow established general engineering practice.

The interdependence fuel and fuel-handling system, the control systems, and some of the internal reactor vessel structure require a more major development program. The Mighty Mouse reactor has a short fuel-operating cycle and this places a premium on refueling time. The presence of the D₂O and the high activity of the irradiated fuel suggest that the refueling must be a machine operation. Such a machine should be capable of unlocking and removing hold-down devices on the fuel assemblies, removing spent fuel assemblies and placing them into proper decay storage positions, reloading the core with new fuel assemblies, and finally, locking the new fuel assemblies in place. Such a system must be built and thoroughly tested. The design of the fuel assemblies and the reactor vessel structure necessary to hold the fuel assemblies is a part of the refueling machine design study.

The control systems need extensive design study. The desired BF₃ shim control system requires study to determine possible response time for control. The system with its circulating pumps, coolers, purification, train, disconnect features, etc., needs to be designed, constructed and tested. Mechanical devices, shock absorbers, blades, disconnect features, etc., for the regulating and safety rods may be adapted to utilize mechanical features developed for other existing reactors.

The final, most important major design problem involves the experimental facilities, beam and irradiation holes. While the type of usage is known, the manner of devising an experiment, the special provisions required in the shield or reactor tank, the cooling problem, and the necessary services required, such as fluids, gases, power supplies, are essential to the ultimate success of the basic concept of the Mighty Mouse Reactor - research.

FIG. 2
CONCEPT OF FLAT PLATE FUEL ASSEMBLY
WITH AXIAL COOLANT FLOW

FIG. 3
CONCEPT OF CURVED PLATE FUEL ELEMENT
ASSEMBLY WITH AXIAL COOLANT FLOW

FIG. 4
SECTION THROUGH D_2O COOLANT
INLET AND OUTLET LINES

FIG. 5
SECTION THROUGH BEAM HOLES
AND GRAPHITE THERMAL COLUMNS

FIG. 6
SECTION THROUGH THERMAL COLUMN AND A BEAM HOLE

FIG. 7
PROPOSED SYSTEM FLOW DIAGRAM

FIG. 8
EMERGENCY COOLER IN REACTOR SHIELD

FIG. 9
REFLECTOR SAVINGS AS A FUNCTION
OF REFLECTOR THICKNESS

RE-7-19703-A

FIG. 10
REFLECTOR SAVINGS VS THERMAL CAPTURE CROSS SECTION.
REFLECTOR THICKNESS = 90 cm

FIG. 11
REFLECTOR SAVINGS VS Al:D₂O VOLUME RATIO.
REFLECTOR THICKNESS = 90 cm

FIG. 12
EFFECT OF ANNULAR THICKNESS OF FUEL
ON PEAK FLUX PER UNIT OF POWER

FIG. 13
NEUTRON FLUX DISTRIBUTION IN BURNED UP CORE

FIG. 14
NORMALIZED RADIAL POWER DISTRIBUTION.
REFLECTOR SAVINGS = 44 cm

FIG. 15
NORMALIZED AXIAL POWER DISTRIBUTION
(SYMMETRICAL ABOUT CENTERLINE).
REFLECTOR SAVINGS = 44 cm

FIG. 16
TEMPERATURE PROFILES ALONG A
0.180 x 2.25 in. CHANNEL

FIG. 17
POWER GENERATED BY FISSION PRODUCT DECAY
AFTER OPERATION AT FULL POWER FOR 5 days

FIG. 18
CONCEPT OF FLAT PLATE ELEMENT
WITH RADIAL COOLANT FLOW

APPENDIX A

SUMMARY OF PRELIMINARY CALCULATIONS OF XENON AND IODINE
CONCENTRATION IN THE MIGHTY MOUSE REACTOR

Problems Treated and Results

The xenon and iodine concentrations, the burn-up of fuel, and the resulting changes in k_{eff} for a prototype Mighty Mouse reactor have been calculated assuming different operating conditions.

The operating conditions considered were as follows:

- I. Constant flux;
- II. Step-wise changes in flux (up and down);
- III. Constant power;
- IV. Step-wise changes in power;
- V. Periodic power.

Under all conditions the constants used were:

$$\begin{array}{llll}
 \lambda^I & = 0.10332/\text{hr} & \sigma_a^I & = 6.7 \times 10^{-24} \text{ cm}^2 & \sigma_f^{U^{235}} & = 580 \times 10^{-24} \\
 \lambda^{Xe} & = 0.07524/\text{hr} & \sigma_a^{Xe} & = 2.7 \times 10^{-18} \text{ cm}^2 & (1+\alpha) \sigma_f^{U^{235}} & = 687 \times 10^{-24} \\
 \gamma^I & = 0.064 & \gamma^{Xe} & = 0.003 & &
 \end{array}$$

For the computation of k_{eff} , reactors with the following parameters were used:

Reactor (A): CP-5 with 3-kg loading:

$$\begin{array}{ll}
 \Sigma_a^{mod} & = 0.001013 \text{ cm}^{-1} & B^2 & = 29.943 \times 10^{-4} \text{ cm}^{-1} \\
 D_s & = 0.84681 \text{ cm} & \tau & = 137.2 \text{ cm}^2 \\
 \text{Volume} & = 195076.19 \text{ cm}^3 & &
 \end{array}$$

Then

$$k_{\text{eff}}(t) = \frac{0.0373513}{0.0270799 + 2.64098 \times 10^{-4}(X_1/X_3) + 1.064276 \times 10^2(X_2/X_3) + (0.0035486/X_3)} \quad (1)$$

where

$$X_1 = \text{iodine concentration/cm}^3$$

$$X_2 = \text{xenon concentration/cm}^3$$

$$X_3 = \text{U}^{235} \text{ concentration/cm}^3$$

If we were to compute the concentrations per reactor, then,

$$k_{\text{eff}}(t) = \frac{1.3793}{1 + (1.00764/X_3)(10^{-24}) + 0.009753(X_1/X_3) + 3.9301 \times 10^3(X_2/X_3)} \quad (2)$$

Reactor (B): 6-kg loading

$$\tau = 170$$

$$B^2 = 0.00147$$

$$D_s = 0.88231$$

$$\Sigma_a^{\text{mod}} = 0.002218$$

$$\text{Volume} = 347.176 \times 10^3 \text{ cm}^3$$

$$k_{\text{eff}}(t) = \frac{1.6201}{1 + (1.77631/X_3)(10^{-24}) + 0.009753(X_1/X_3) + 3.9301 \times 10^3(X_2/X_3)} \quad (3)$$

where the concentrations are given per reactor.

Reactor (C): Same as Reactor (B) with a loading of 4 kg.

$$\tau = 170$$

$$B^2 = 0.00147$$

$$D_s = 0.88231$$

$$\Sigma_a^{\text{mod}} = 0.002219$$

$$\text{Volume} = 347.176 \times 10^3 \text{ cm}^3$$

and k_{eff} is given by Eq. (3).

For Type I operation two fluxes were considered:

$$\Phi = 5 \times 10^{15} \text{ n/(cm}^2\text{)(sec)} = 1.8 \times 10^{19} \text{ n/(cm}^2\text{)(hr)}$$

and

$$\Phi = 10^{15} \text{ n}/(\text{cm}^2)(\text{sec}) = 0.36 \times 10^{19} \text{ n}/\text{cm}^3/\text{hr}$$

were used in conjunction with reactor (A) and Eq. (1). The results are given in Figs. 19 and 20. It appears that in the case $\Phi = 5 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ xenon peaks at about 22 hr and in the case of $\Phi = 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$, at 36.7 hr.

The solution for the concentrations under condition II permits the determination of the behaviour of k_{eff} when the flux is suddenly changed. Again, Reactor (A) was used as a prototype. This reactor was set to operate at a flux $\Phi = 5 \times 10^{14} \text{ n}/(\text{cm}^2)(\text{sec})$ for 24 hr, then the flux was raised and held at $10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ for another 24 hr, at the end of which the flux was raised to and operated at $5 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ for 48 hr. The results for this type of operation are given in Fig. 21.

Figure 22 shows the effect of starting at $\Phi = 5 \times 10^{14} \text{ n}/(\text{cm}^2)(\text{sec})$ and after 24 hr raising the flux to $5 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ and operating at that level for 72 hr.

As is evident from Figs. 21 and 22, the increase in k_{eff} due to the considerable increase in flux does not appear to be so excessive as to cause a serious problem of control.

The possibility of a reactor shutdown due to xenon when the flux drops was considered. For this purpose Reactor (A) was considered to have concentrations of xenon, iodine, and fuel, and a k_{eff} which would result if the reactor was operated at $\Phi = 5 \times 10^{14} \text{ n}/(\text{cm}^2)(\text{sec})$ for 24 hr and then operated at $\Phi = 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ for 12 hr. At the end of these 36 hr the flux goes instantaneously to (a) $\Phi = 2 \times 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$ and (b) $\Phi = 10^{15} \text{ n}/(\text{cm}^2)(\text{sec})$. The results are given in Figs. 23 and 24. Although reactivity is lost, it is not lost at an unmanageable rate. After the xenon peak is reached, the rate of decrease in the concentration follows a curve almost parallel to the iodine curve. In this case the loss in reactivity may not be easily compensated. The xenon peak is reached very quickly after the drop of flux.

To treat the constant power cases, the method outlined on page 62 was used. Reactors (A), (B) and (C) were the prototypes. To study the build-up of the xenon and iodine, as well as the fuel burnup, the reactors were considered to operate at 300 megawatts. It was found simpler in these cases to compute the concentrations per reactor instead of per cm^3 . The corrected formulas for k_{eff} were then used. The results are given in Figs. 25, 26, and 27.

As in the constant flux operation, again there is an initial drop in k_{eff} at a substantial rate. As the xenon and iodine approach their peaks, the rate of drop in k_{eff} becomes smaller and fairly constant, being mainly due to the depletion of the fuel. For the same period of time the different reactors show a different total change in k_{eff} and again the fuel burnup over this period of time accounts for the difference.

It is of interest to determine the behaviour of the concentrations and k_{eff} if the power is changed instantaneously to a lower level. For reactor (A) the power level was changed to 120 megawatts following operation at 300 mw for 10 hr. Also, for reactor (A) the power was dropped to 60 mw after 10 hr operation at 300 mw. For reactor (B) the power level was dropped to 60 mw after 20 hr operation at 300 mw. The results are given in Figs. 28, 29, and 30. The drop in power by the same factor as the drop in flux does not change k_{eff} quite as much as in the constant flux case. The rate of change is also smaller and seems to be more uniform.

The effect of power oscillation was computed for the case of Reactor (B). The method used is outlined on page 63. The power was assumed to be of the form:

$$\sigma_f X_3 \Phi = p_0 + p \sin wt \quad ,$$

where

$$p_0 = 300 \text{ mw}$$

$$p = 0.1 p_0$$

$$w = 1/2 \text{ cycle/hr} \quad .$$

The initial values were:

$$X_1 = 1.8521 \times 10^{-22}$$

$$X_2 = 0.79052 \times 10^{-20}$$

$$X_3 = 6.8782 \times 10^{-24} \quad .$$

The results (Fig. 31) indicated that the total change in k_{eff} is about 1% for a power swing of 10%. It is not surprising that k_{eff} is 180° out of phase with the xenon. After a fairly short time, the xenon concentration becomes periodic and follows the shifted power.

Methods and Derivations

The equations describing the concentration of xenon and iodine as functions of time, including fuel burnup, are solved for the following cases:

I. Constant Flux	$\Phi = \text{constant} = \mu_1$
II. Constant Power	$\sigma_f X_3 \Phi = \text{constant} = \mu$
III. Periodic Power	$\sigma_f X_3 \Phi = p_0 + p \sin wt$

In every case the equations are:

$$\frac{dX_1}{dt} = a_1 X_1 + a_3 \sigma_f X_3 \Phi \quad (1)$$

$$\frac{dX_2}{dt} = b_1 X_1 + (b_2 + b_4 \Phi) X_2 + b_3 \sigma_f X_3 \Phi \quad (2)$$

$$\frac{dX_3}{dt} = c_3 \sigma_f X_3 \Phi \quad (3)$$

where

$X_1(0)$ = initial iodine concentration

$X_2(0)$ = initial xenon concentration

$X_3(0)$ = initial fuel concentration

$X_1(t)$ = iodine concentration

$X_2(t)$ = xenon concentration

$X_3(t)$ = fuel concentration

a_3 = iodine yield

$-b_2$ = xenon decay constant

b_3 = direct xenon yield

$-b_4$ = xenon absorption cross section

$-a_1 = b_1$ = iodine decay constant

$c_3 = -1.183$

For Case I the solutions are:

$$X_1(t) = \left[\frac{a_3 \sigma_f \mu_1 X_3(0)}{a_1 - c_3 \sigma_f \mu_1} + X_1(0) \right] \exp(a_1 t) - \frac{a_3 \sigma_f \mu_1 X_3(0)}{a_1 - c_3 \sigma_f \mu_1} \exp(c_3 \sigma_f \mu_1 t) - \alpha \exp(c_3 \sigma_f \mu_1 t)$$

$$= \beta \exp(a_1 t)$$

$$X_2(t) = \frac{b_1 \beta}{a_1 - b_2 - b_4 \mu_1} \exp(a_1 t) - \left[\frac{b_1 \beta}{a_1 - b_2 - b_4 \mu} + \frac{b_3 \sigma_f \mu_1 X_3(0) - \alpha b_1}{c_3 \sigma_f \mu_1 - b_2 - b_4 \mu} - X_2(0) \right] \exp(b_2 + b_4 \mu_1) t$$

$$+ \frac{b_3 \sigma_f \mu_1 - \alpha b_1}{c_3 \sigma_f \mu_1 - b_2 - b_4 \mu_1} \exp(c_3 \sigma_f \mu_1 t)$$

$$X_3(t) = X_3(0) \exp(c_3 \sigma_f \mu_1 t)$$

For Case II Eqs. (1), (2), and (3) become

$$\frac{dX_1}{dt} = a_1 X_1 + a_3 \mu \quad (4)$$

$$\frac{dX_2}{dt} = b_1 X_1 + \left(b_2 + \frac{b_4 \mu}{\sigma_f X_3} \right) X_2 + b_3 \mu \quad (5)$$

$$\frac{dX_3}{dt} = c_3 \mu \quad (6)$$

Now

$$X_3(t) = c_3 \mu t + X_3(0) ,$$

and, for a given interval, $\Delta t = t_j - t_{j-1}$, in which $X_3(t)$ does not change much. We may define

$$\lambda_j = \frac{2b_4 \mu}{\sigma_f X_3(t_j) + X_3(t_{j-1})} .$$

This is used in Eqs. (1) to (3). The solutions are

$$X_1(t_j) = \left[X_1(0) + \frac{a_3 \mu}{a_1} \right] \exp(a_1 t_j) - \frac{a_3 \mu}{a_1}$$

$$= A \exp(a_1 t_j) - \beta_1$$

$$X_3(t_j) = X_3(t_{j-1}) + c_3 \mu \Delta t$$

$$\begin{aligned}
X_2(t_j) = & X_2(t_{j-1}) \exp (b_2 + \lambda_j) \Delta t + \frac{\mu(b_3 + a_3)}{-b_2 - \lambda_j} \left\{ 1 - \exp [(b_2 + \lambda_j) \Delta t] \right\} \\
& + \frac{b_1 A}{a_1 - b_2 - \lambda_j} \left\{ \exp (a_1 t_j) - \exp [a_1 t_{j-1} + (b_2 + \lambda_j) \Delta t] \right\} .
\end{aligned}$$

For the periodic power case we have

$$\begin{aligned}
\frac{dX_1}{dt} &= a_1 X_1 + a_3 (p_0 + p \sin wt) \\
\frac{dX_2}{dt} &= b_1 X_1 + \left(b_2 + \frac{b_4}{\sigma_f c_3} \frac{1}{X_3} \frac{dX_3}{dt} \right) X_2 + b_3 (p_0 + p \sin wt) \\
\frac{dX_3}{dt} &= c_3 (p_0 + p \sin wt) .
\end{aligned}$$

For Δt sufficiently small $\frac{d \ln X_3}{dt}$ is of small variation in the interval $t_{j-1} \leq t \leq t_j$; therefore, defining:

$$\Delta_j = \frac{b_4}{c_3 \sigma_f} \frac{1}{\Delta t} \int_{t_{j-1}}^{t_j} \frac{1}{X_3} \frac{dX_3}{dt} dt = \frac{b_4}{c_3 \sigma_f} \frac{1}{\Delta t} \log \frac{X_3(t_j)}{X_3(t_{j-1})} ,$$

we obtain

$$\begin{aligned}
X_1(t_j) = & \left[X_1(0) + \frac{a_3 p_0}{a_1} + \frac{a_3 w p}{a_1^2 + w^2} \right] \exp [a_1 t_j] \\
& - \left[\frac{a_3 p_0}{a_1} + \frac{a_1 a_3 p}{a_1^2 + w^2} \sin wt_j + \frac{a_3 w p}{a_1^2 + w^2} \cos wt_j \right] \\
X_3(t_j) = & c_3 (p_0 t_j - \frac{p}{w} \cos wt_j) + c_3 \frac{p}{w} + X_3(0)
\end{aligned}$$

$$\begin{aligned}
X_2(t_j) = & X_2(t_{j-1}) \exp [(b_2 + \Delta_j) \Delta t] \\
& + \frac{\alpha_1}{a_1 - b_2 - \Delta_j} \left\{ \exp a_1 t_j - \exp [a_1 t_{j-1} + (b_2 + \Delta_j) \Delta t] \right\} \\
& - \frac{\alpha_2}{b_2 + \Delta_j} \left\{ 1 - \exp [(b_2 + \Delta_j) \Delta t] \right\} \\
& + \frac{\sin w t_j}{w^2 + (b_2 + \Delta_j)^2} \left\{ \alpha_4 w - \alpha_3 (b_2 + \Delta_j) \right\} \\
& + \frac{\exp [(b_2 + \Delta_j) \Delta t] \sin w t_{j-1}}{w^2 + (b_2 + \Delta_j)^2} \left\{ \alpha_3 (b_2 + \Delta_j) - \alpha_4 w \right\} \\
& - \frac{\cos w t_j}{w^2 + (b_2 + \Delta_j)^2} \left\{ \alpha_4 (b_2 + \Delta_j) + w \alpha_3 \right\} \\
& + \frac{\exp [(b_2 + \Delta_j) \Delta t] \cos w t_{j-1}}{w^2 + (b_2 + \Delta_j)^2} \left\{ \alpha_4 (b_2 + \Delta_j) + w \alpha_3 \right\} ,
\end{aligned}$$

where

$$\begin{aligned}
\alpha_1 &= b_1 \left[X_1(0) + \frac{a_3 p_0}{a_1} + \frac{a_3 w p}{a_1^2 + w^2} \right] \\
\alpha_2 &= \left[b_3 p_0 - \frac{b_1 a_3 p_0}{a_1} \right] \\
\alpha_3 &= \left[b_3 p - \frac{a_1 a_3 b_1}{a_1^2 + w^2} \right] \\
\alpha_4 &= - \frac{a_3 w b_1 p}{a_1^2 + w^2} .
\end{aligned}$$

In a study of the time behavior of the xenon concentration for a constant power operation of a reactor, D. H. Shaftman, considering the magnitudes of the constants involved, derived a simple relation for the time behavior of the xenon:

$$\frac{\sum_a^{Xe}(t)}{\sum_f^{fuel}(t)} \approx \gamma(Xe) + \gamma(I) - \gamma(I) \exp(-\lambda^I t) \quad (t > 0) \quad . \quad (10)$$

If in the formula for $X_2(t_j)$ for constant power operation all terms involving $\exp(b_2 + \lambda_j)t$ are neglected, being insignificant in the cases considered we obtain

$$X_2(t_j) = \frac{\mu(b_3 + a_3)}{-b_2 - \lambda_j} + \frac{b_1 a_3 \mu}{a_1(a_1 - b_2 - \lambda_j)} \exp(a_1 t_j) \quad (t > 0) \quad , \quad (11)$$

which is a slightly more exact version of Eq. (7).

Values obtained by Eqs. (10) and (11) differ at most by 3% in the cases considered.

TIME	$X_1 (10^2)$	$X_2 (10^5)$	X_3	k_{eff}
0	0	0	1.00000	1.21949
0.15	0.00994	0.08246	0.99815	1.21574
1.5	0.09197	0.25600	0.98162	1.20594
5.0	0.25233	0.59450	0.94004	1.18461
8.0	0.34399	0.78740	0.90581	1.17002
10.0	0.38773	0.87910	0.88368	1.16158
15.0	0.45428	1.01740	0.83069	1.14375
20.0	0.48062	1.07030	0.78089	1.12897
21.0	0.48269	1.07410	0.77128	1.12624
22.0	0.48398	1.07640	0.76182	1.12357
23.0	0.48453	1.07700	0.75246	1.12096
25.0	0.48375	1.07410	0.73406	1.11586
30.0	0.47381	1.05027	0.69005	1.10363
35.0	0.45677	1.01148	0.64868	1.09177
40.0	0.43617	0.96530	0.60979	1.07996

FIG. 19
IODINE, XENON, URANIUM CONCENTRATIONS
AND k_{eff} AS FUNCTIONS OF TIME IN
REACTOR "A". ($\Phi = 5 \times 10^{15}$)

TIME	$X_1 (10^3)$	$X_2 (10^5)$	X_3	k_{eff}
0	0	0	1.00000	1.21949
0.5	0.06512	0.09735	0.99880	1.21521
1.0	0.12678	0.18449	0.99753	1.21136
3.0	0.34339	0.41513	0.99260	1.20100
5.0	0.51831	0.60137	0.98770	1.19254
7.0	0.65942	0.75155	0.98283	1.18559
10.0	0.82116	0.92360	0.97557	1.17734
12.0	0.90276	1.10134	0.97071	1.17296
15.0	0.99553	1.10890	0.96358	1.16768
18.0	1.06107	1.17846	0.95645	1.16353
20.0	1.09337	1.21261	0.95176	1.16125
25.0	1.14552	1.26763	0.94003	1.15673
30.0	1.17062	1.29381	0.92849	1.15335
35.0	1.17958	1.30280	0.91707	1.15063
36.0	1.18008	1.30323	0.91480	1.15014
36.7	1.18023	1.30331	0.91322	1.14980
37.0	1.18024	1.30327	0.91255	1.14966
38.0	1.18011	1.30304	0.91030	1.14920
45.0	1.17286	1.29453	0.89468	1.14622
50.0	1.16337	1.28387	0.88366	1.14422

FIG. 20
IODINE, XENON, URANIUM CONCENTRATIONS AND k_{eff}
AS FUNCTIONS OF TIME IN REACTOR "A". ($\phi = 10^{15}$)

TIME	$X_1 (10^3)$	$X_2 (10^5)$	X_3	k_{eff}
$\phi = 5 \times 10^{14}$				
0	0	0	1.00000	1.21949
2	0.12057	0.29228	0.99753	1.20686
4	0.21833	0.50119	0.99507	1.19784
8	0.36169	0.80755	0.99017	1.18456
12	0.45544	1.00785	0.98529	1.17567
16	0.51639	1.13801	0.98043	1.16961
20	0.55564	1.22178	0.97560	1.16540
$\phi = 10^{15}$				
24	0.58055	1.27487	0.97079	1.16239
26	0.70597	0.80033	0.96599	1.18068
28	0.80683	0.90761	0.96124	1.17543
30	0.88771	0.99359	0.95649	1.17104
32	0.95234	1.06223	0.95176	1.16736
36	1.04447	1.16002	0.94239	1.16157
40	1.10135	1.22019	0.93312	1.15729
44	1.13492	1.25553	0.92394	1.15399
$\phi = 5 \times 10^{15}$				
48	1.15313	1.27449	0.91484	1.15133
52	2.71348	0.63070	0.87069	1.16989
56	3.65150	0.82765	0.82867	1.15188
60	4.18233	0.93811	0.78867	1.13714
64	4.44825	0.99233	0.75062	1.12450
68	4.54293	1.01024	0.71439	1.11317
72	4.52832	1.00501	0.67991	1.10265
76	4.44521	0.98532	0.64710	1.09259
80	4.32021	0.95682	0.61587	1.08274
84	4.17090	0.92323	0.58615	1.07294
88	4.00881	0.88701	0.55786	1.06308
92	3.84126	0.84971	0.53094	1.05308
96	3.67300	0.81236	0.50531	1.04288

FIG. 21
MATERIAL CONCENTRATIONS AND k_{eff} AS
FUNCTIONS OF TIME WITH STEP-WISE
VARIATIONS IN POWER OF REACTOR "A"

TIME	$X_1 (10^3)$	$X_2 (10^5)$	X_3	k_{eff}
$\Phi = 5 \times 10^{14}$				
0	0	0	1.00000	1.21949
2	0.12057	0.29223	0.99753	1.20686
4	0.21833	0.50119	0.99507	1.19784
8	0.36169	0.80755	0.99017	1.18456
12	0.45544	1.00785	0.98529	1.17567
16	0.51639	1.13801	0.98043	1.16961
20	0.55564	1.22178	0.97560	1.16540
$\Phi = 5 \times 10^{15}$				
24	0.58055	1.27487	0.97079	1.16239
30	3.09720	0.71420	0.90137	1.17232
36	4.25198	0.95589	0.83691	1.14797
42	4.68839	1.04504	0.77707	1.12914
48	4.75139	1.05503	0.72149	1.11309
54	4.62588	1.02515	0.66989	1.09832
60	4.41040	0.97638	0.62199	1.08398
66	4.15706	0.91976	0.57751	1.06958
72	3.89312	0.86108	0.53622	1.05486
78	3.63273	0.80333	0.49788	1.03966
84	3.38253	0.74792	0.46226	1.02387
90	3.14586	0.69554	0.42921	1.00747
96	2.92371	0.64641	0.39852	0.99042

FIG. 22
EFFECT OF A TENFOLD INCREASE IN POWER
ON CONCENTRATIONS AND k_{eff} IN REACTOR "A"

TIME	$X_1 (10^3)$	$X_2 (10^5)$	X_3	k_{eff}
0	4.25198	0.95589	0.83691	1.14797
0.05	4.24122	1.79437	0.83670	1.11149
0.1	4.23052	2.10683	0.83650	1.09844
0.3	4.18823	2.27286	0.83567	1.09143
0.5	4.14682	2.25466	0.83487	1.09195
1.0	4.04657	2.20119	0.83278	1.09357
2.0	3.86028	2.10180	0.82867	1.09654
4	3.53776	1.92967	0.82052	1.10152
6	3.27152	1.78751	0.81244	1.10538
8	3.05111	1.66974	0.80444	1.10831
10	2.86800	1.57183	0.79652	1.11045
12	2.71527	1.49016	0.78868	1.11194

FIG. 23
EFFECT OF SUDDEN DECREASE IN FLUX LEVEL ON
CONCENTRATIONS AND k_{eff} IN REACTOR "A".
FINAL FLUX LEVEL: $\Phi = 2 \times 10^{15}$

TIME	$X_1 (10^3)$	$X_2 (10^5)$	X_3	k_{eff}
0	4.25198	0.9559	0.83691	1.14797
0.05	4.23565	3.8392	0.83681	1.03167
0.1	4.21940	4.0997	0.83670	1.02236
0.2	4.18715	4.3399	0.83650	1.01357
0.3	4.15523	4.4065	0.83629	1.01135
0.5	4.09236	4.3947	0.83588	1.01162
1	3.94072	4.2422	0.83484	1.01635
2	3.65975	3.9426	0.83278	1.02668
4	3.17709	3.4279	0.82867	1.04433
6	2.78355	3.0082	0.82458	1.05910
8	2.46249	2.6656	0.82051	1.07137
10	2.20039	2.3860	0.81647	1.08148
12	1.98624	2.1576	0.81244	1.08942

FIG. 24
EFFECT OF SUDDEN DECREASE IN FLUX
LEVEL ON CONCENTRATIONS AND k_{eff} IN
REACTOR "A". FINAL FLUX LEVEL: $\Phi = 10^5$

TIME	X_1 (10 ⁻²²)	X_2 (10 ⁻¹⁹)	X_3 (10 ⁻²⁴)	k_{eff}
0	0	0	7.68896	1.21949
5	0.8556	4.65	7.48628	1.19006
10	1.3660	6.797	7.28360	1.17384
15	1.6705	8.204	7.08092	1.16116
20	1.8521	8.796	6.87824	1.15251
25	1.9605	9.169	6.67556	1.14469
30	2.0251	9.011	6.47287	1.13953
40	2.0867	8.712	6.0751	1.12823
50	2.1086	8.225	5.66223	1.11677
60	2.1164	7.677	5.25687	1.10422
70	2.1192	7.107	4.85150	1.09009
80	2.1202	6.529	4.44614	1.07389
90	2.1205	5.949	4.04078	1.05509

FIG. 25
CONCENTRATIONS AND k_{eff} AS FUNCTIONS OF TIME
FOR REACTOR "A" AT CONSTANT POWER OF 300 mw

TIME	$X_1 (10^{-22})$	$X_2 (10^{-20})$	$X_3 (10^{-25})$	k_{eff}
0	0	0	1.5378	1.45231
5	0.8556	0.9207	1.5175	1.41999
10	1.3660	1.3866	1.4972	1.40260
15	1.6705	1.6333	1.4770	1.39213
20	1.8521	1.7524	1.4567	1.38559
25	1.9605	1.7973	1.4364	1.38131
30	2.0251	1.7991	1.4162	1.37835
40	2.0867	1.7383	1.3756	1.37443
50	2.1086	1.6410	1.3351	1.37135
60	2.1164	1.5316	1.2946	1.36862
70	2.1192	1.4181	1.2540	1.36587
80	2.1202	1.3031	1.2135	1.36301
90	2.1205	1.1876	1.1730	1.35998
100	2.1206	1.0718	1.1324	1.35676
110	2.1207	0.9560	1.0919	1.35332
120	2.1207	0.8400	1.0514	1.34964

FIG. 26
BEHAVIOR OF REACTOR "B" AT CONSTANT
POWER OF 300 mw

TIME	$X_1 (10^{-22})$	$X_2 (10^{-20})$	$X_3 (10^{-25})$	k_{eff}
0	0	0	1.02519	1.3808
5	0.8556	0.6200	1.0049	1.3490
10	1.3660	0.9375	0.9847	1.3303
15	1.6705	1.1113	0.9644	1.3177
20	1.8521	1.2009	0.9441	1.3085
25	1.9605	1.2412	0.9239	1.3012
30	2.0251	1.2529	0.9036	1.2949
40	2.0867	1.2324	0.8630	1.2838
50	2.1086	1.1872	0.8225	1.2729
60	2.1164	1.1336	0.7820	1.2616
70	2.1192	1.0772	0.7414	1.2494
80	2.1160	1.0197	0.7009	1.2361
90	2.1205	0.9561	0.6604	1.2219
100	2.1206	0.9040	0.6198	1.2055
110	2.1207	0.8460	0.5793	1.1877
120	2.1207	0.7880	0.5388	1.1679

FIG. 27
BEHAVIOR OF REACTOR "C" AT CONSTANT POWER OF 300 mw

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TIME	$X_1 (10^{-22})$	$X_2 (10^{-20})$	$X_3 (10^{-24})$	k_{eff}
0	1.3660	0.6689	7.2836	1.1744
0.5	1.3633	1.0167	7.2829	1.1559
0.10	1.3607	1.2392	7.2820	1.1444
0.15	1.3580	1.3811	7.2813	1.1371
0.20	1.3554	1.4707	7.2804	1.1326
0.25	1.3528	1.5275	7.2795	1.1297
0.30	1.3502	1.5628	7.2787	1.1280
0.35	1.3476	1.5843	7.2780	1.1269
0.40	1.3450	1.5969	7.2772	1.1262
0.45	1.3424	1.6037	7.2763	1.1258
0.50	1.3399	1.6067	7.2755	1.1257
1	1.3152	1.7873	7.2674	1.1165
2	1.2694	1.5280	7.2512	1.1289
5	1.1571	1.3865	7.2025	1.1347

FIG. 28
BEHAVIOR OF REACTOR "A" AFTER DROP
IN POWER FROM 300 mw TO 120 mw

TIME	$X_1 (10^{-22})$	$X_2 (10^{-20})$	$X_3 (10^{-24})$	k_{eff}
0	1.3660	0.6689	7.2836	1.1744
0.05	1.3612	1.1736	7.2832	1.1480
0.10	1.3563	1.5756	7.2829	1.1274
0.15	1.3515	1.8941	7.2824	1.1118
0.20	1.3467	2.1478	7.2820	1.0997
0.25	1.3420	2.3486	7.2816	1.0903
0.30	1.3373	2.5070	7.2813	1.0829
0.35	1.3326	2.6316	7.2808	1.0772
0.40	1.3279	2.7314	7.2804	1.0727
0.45	1.3231	2.8113	7.2800	1.0691
0.50	1.3186	2.8684	7.2795	1.0665
1	1.2736	3.0036	7.2755	1.0604
2	1.1902	2.8282	7.2674	1.0679
3	1.1150	2.6464	7.2593	1.0758

FIG. 29
BEHAVIOR OF REACTOR "A" AFTER DROP IN
POWER FROM 300 mw TO 60 mw

TIME	$X_1 (10^{-22})$	$X_2 (10^{-20})$	$X_3 (10^{-25})$	k_{eff}
0	1.8521	1.7524	1.4567	1.3855
0.05	1.8448	2.4772	1.4567	1.3628
0.1	1.8315	3.1210	1.4566	1.3416
0.15	1.8302	3.6960	1.4566	1.3261
0.2	1.8229	4.2017	1.4566	1.3114
0.25	1.8157	4.6503	1.4565	1.2987
0.3	1.8085	5.0474	1.4565	1.2876
0.35	1.8014	5.3984	1.4564	1.2780
0.4	1.7943	5.7084	1.4564	1.2696
0.45	1.7871	5.9810	1.4564	1.2623
0.5	1.7802	6.2217	1.4563	1.2560
1.0	1.7120	7.4402	1.4560	1.2247
1.5	1.6471	7.6291	1.4555	1.2199
2.0	1.5855	7.4928	1.4551	1.2232
2.5	1.5271	7.2617	1.4547	1.2289

FIG. 30
BEHAVIOR OF REACTOR "B" AFTER DROP IN
POWER FROM 300 mw TO 60 mw

TIME	$X_1 (10^{-22})$	$X_2 (10^{-20})$	$X_3 (10^{-24})$	k_{eff}
0	1.8521	0.79052	6.8782	1.1574
0.25	1.8610	0.8412	6.8677	1.1543
0.5	1.8725	0.8057	6.8567	1.1560
0.75	1.8837	0.8082	6.8456	1.1556
1.0	1.8917	0.8486	6.8351	1.1530
1.25	1.8956	0.9114	6.8254	1.1493
1.5	1.8964	0.9615	6.8161	1.1463
1.75	1.8973	0.9616	6.8069	1.1460
2.0	1.9001	0.9125	6.7972	1.1484
2.25	1.9086	0.8521	6.7867	1.1515
2.5	1.9189	0.8150	6.7756	1.1532
2.75	1.9289	0.8169	6.7646	1.1528
3.0	1.9358	0.8589	6.7541	1.1502
3.25	1.9385	0.9198	6.7443	1.1465
3.5	1.9382	0.9702	6.7351	1.1435
3.75	1.9380	0.9698	6.7259	1.1432
4.0	1.9407	0.9196	6.7161	1.1457

FIG. 3I
EFFECT OF POWER OSCILLATION ON REACTOR "B"

APPENDIX B

A TWO-GROUP ITERATION METHOD FOR ANNULAR CYLINDERS

In the course of work on Argonaut,⁸ a low-power research reactor, it became necessary to investigate the critical properties of a homogeneous annular region reflected externally and internally. An iterative technique, which was characterized by ease of application and rapidity of convergence, was evolved to solve these problems. While the idea of an iterative technique is by no means new,⁹ the formalism used here is, to our knowledge, unique and well suited to this particular problem.

Although there are several ways of determining the critical properties of a homogeneous annulus, they are each confronted with certain difficulties which make solution laborious. Analytical methods enable one to satisfy the boundary conditions automatically, but the considerable algebraic manipulation of Bessel functions of nearly the same argument leads to an excessive sensitivity to minor errors in setting up the work. Straight-forward solution by numerical integration over the entire volume is complicated by the problem of crossing boundaries. Hence, a technique has been developed which utilizes the results of analysis to find the proper boundary conditions at the edges of the annulus and then uses a numerical iteration scheme in the interior.

The two-group equations are reduced to four first-order equations by the introduction of the current as an independent variable. The interior of the annulus is divided into a number of equal intervals and it is assumed that the current and the flux vary linearly in the interior of each interval and are continuous at the interval end-points. The equations can then be integrated over an interval to yield easily manipulated algebraic equations which give the values of the flux at the point $i + 1$ in terms of the values at the point i . These equations have the form:

$$J_{i+1} = J_i + \lambda_i (\alpha_i J_i - \beta_i \phi_i + q_i)$$

$$\phi_{i+1} = \phi_i - \eta (J_i + J_{i+1})$$

where λ_i , α_i , β_i are functions of r .

⁸D. H. Lennox, C. N. Kelber, "Summary Report on the Hazards of the Argonaut Reactor," ANL-5647 (November, 1955).

⁹L. Noderer, "Some Methods of Solving Multi-Group Pile Equations," ORNL-291 (March 2, 1949) (Declassified January 15, 1955).

The relation between the flux and the current in each group at the inner and outer boundary has the form

$$J = m \phi + b ,$$

where m is a function of reflector constants only, and b , for the thermal flux, is such a function times the fast flux at the boundary. For fast flux in problems with non-multiplying boundaries, b is zero. For complicated internal and external reflectors, variants of the matrix method¹⁰ may be used to determine the coefficients m and b in the boundary conditions.

To start an iteration, an initial shape of the slow flux is assumed and an initial value of the fast flux is guessed, the analytical boundary conditions furnishing the correct corresponding value of the current. The iteration is carried out to the outer boundary. Sufficient homogeneous solution (i.e., the solutions of the equations with $q \equiv 0$ and $b = 0$ at the internal boundary) is added to satisfy the analytical boundary condition there. The flux thus obtained furnishes a source term for the slow flux. A similar process yields a new estimate of the slow flux. A comparison of the weighted averages of the initial and final slow flux distributions gives an estimate of the reactivity of the system.

Our experience has been that one iteration takes about four hours of hand computation and yields the reactivity with an accuracy of about 1%, even for rather bad guesses of the flux shape. This compares favorably with the same problems done by purely analytical or purely numerical methods.

A complete exposition of this method will be published as an Argonne topical report.

¹⁰D. Kurath, B. I. Spinrad, "Computation Forms for Solution of Critical Problems by Two-Group Diffusion Theory," ANL-4352 (March, 1952).

APPENDIX C

SPECIFICATIONS FOR BURNABLE POISONS

Approximate calculations were made to determine the concentrations of combinations of various burnable poisons required for control and operation of the reactor.

The calculations were divided into two parts. In the first part three absorbers were used to compensate for the xenon buildup. In the second part two absorbers were used to maintain the proper value of the thermal utilization factor (f) for criticality.

The xenon compensation was calculated as follows:

Let α_i be the microscopic absorption cross section for the poisons:

$$\alpha_1 = \sigma_a^{\text{Cd}^{113}}$$

$$\alpha_2 = \sigma_a^{\text{B}^{10}}$$

$$\alpha_3 = \sigma_a^{\text{Li}^6}$$

Assuming the reactor to be operating at a constant power (μ), the concentrations of the poisons (A_i) are determined by the equation:

$$A_i(t) = A_i(0) \left[1 + \frac{c_3 \mu}{X_3(0)} t \right]^{(\alpha_i / c_3 \sigma_f)} \quad (1)$$

where

$$c_3 = -1.183$$

$$\sigma_f = \text{microscopic fission cross section of the fuel}$$

$$X_3(0) = \text{initial concentration of fuel}$$

$$t = \text{time, hr}$$

At time t the fuel concentration is

$$X_3(t) = X_3(0) \left[1 + \frac{c_3 \mu}{X_3(0)} t \right] \quad (2)$$

and the iodine concentration is

$$X_1(t) = 0.61943\mu [1 - \exp(-0.10332t)] \quad (3)$$

For $0 \leq t \leq t_1$, where t_1 is the time at which the xenon concentration, $X_2(t)$, is a maximum,

$$\sigma_1 X_1(t) + \sigma_2 X_2(t) + \sum_{i=1}^3 \alpha_i A_i(t) = \text{constant} = \xi \quad (4)$$

Upon differentiating Eq. (4) and using the equation for xenon concentration, one obtains

$$\begin{aligned} X_1(t) \left[\sigma_2 b_1 + \sigma_1 a_1 - \sigma_1 b_2 + \frac{\sigma_1 \sigma_2 \mu}{\sigma_f X_3(t)} \right] + \mu(\sigma_1 a_3 + \sigma_2 b_3) + \xi \left[b_2 - \frac{\sigma_2 \mu}{\sigma_f X_3(t)} \right] \\ = \sum_{i=1}^3 A_i(t) \left(\frac{\alpha_i 2\mu}{\sigma_f X_3} + \alpha_i b_2 - \frac{\alpha_i \sigma_2 \mu}{\sigma_f X_3} \right) \end{aligned} \quad (5)$$

where

σ_1 = corrected microscopic iodine cross section

$$= (0.8862)(6.7) 10^{-24} = 5.94 \times 10^{-24} \text{ cm}$$

σ_2 = corrected microscopic xenon cross section

$$= 2.39 \times 10^{-18}$$

a_3 = iodine yield = 0.064

b_3 = direct xenon yield = 0.003

$-b_2$ = xenon decay constant/hr = 0.07524

b_1 = iodine decay constant/hr = 0.10332

$a_1 = -b_1$

From Appendix A it appears that $t_1 \approx 25$ hr for the contemplated type of operation. Using:

$$\mu = 2.853 \times 10^{22} \text{ fissions/hr} = 250 \text{ megawatts}$$

$$X_3(0) = 8.050318 \times 10^{24} \text{ atoms/reactor} = 3141 \text{ gm } U^{235}/\text{reactor}$$

$$\alpha_1 = 23462.9 \times 10^{-24}$$

$$\alpha_2 = 3554 \times 10^{-24}$$

$$\alpha_3 = 837.5 \times 10^{-24}$$

$$\sigma_f = 499.7464 \times 10^{-24}$$

in Eqs. (1), (2), (3), (5) for $t = 0, 5, 15, 25$, respectively, we obtain

$$A_1(0) = \text{Cd}^{113} \text{ concentration/reactor} = 9.2118 \times 10^{22} \text{ atoms/reactor}$$

$$A_2(0) = \text{B}^{10} \text{ concentration/reactor} = 1.01311 \times 10^{22} \text{ atoms/reactor}$$

$$A_3(0) = \text{Li}^6 \text{ concentration/reactor} = 0.061985 \times 10^{24} \text{ atoms/reactor}$$

$$\xi = 287.39$$

Criticality calculations indicate that the values of f needed to operate at the given power level are:

$$f(\text{at } t = 0) = 0.6381$$

$$f(\text{at } t = 120 \text{ hr}) = 0.73965$$

from

$$f = \frac{(1 + \alpha) \sigma_f X_3(t)}{(1 + \alpha) \sigma_f X_3(t) + \sigma_1 X_1(t) + \sigma_2 X_2(t) + \sigma^{\text{mod}} + \sigma^{\text{fp}} + \sum_{i=2}^3 \alpha_i A_i + S} \quad (6)$$

Knowing f at $t = 0$, $t = 120$, and all other terms in Eq. (6) except $\sum_{i=2}^3 \alpha_i A_i$, the additional amounts of boron and lithium are determined. At $t = 0$ we obtain:

$$\alpha_2 A_2' + \alpha_3 A_3' = 2106.519/\text{reactor} \quad , \quad (7)$$

and for $t = 120$,

$$0.014933 \alpha_2 A_2' + 0.323642 \alpha_3 A_3' = 201.9868/\text{reactor} \quad , \quad (8)$$

if we use

$$\sigma_2 X_2(120) = 136.4571$$

$$\sigma_1 X_1(120) = 0.1050$$

$$\sigma_{fp} = 171.18 = \text{fission product poison}$$

$$\sigma_{mod} = 305.3625$$

$$(1 + \alpha) \sigma_f X_3(120) = 2364.92$$

and the poison contributed from xenon compensations is 17.3387. Solving Eqs. (7) and (8):

$$A_2' = 0.437288 \times 10^{24} \text{ atoms/reactor}$$

$$A_3' = 0.659580 \times 10^{24} \text{ atoms/reactor} .$$

Consequently, the total amounts of burnable poison are:

$$A_1(0) = \text{Cd}^{113} \text{ concentrations/reactor} = 9.2118 \times 10^{22} \text{ atoms/reactor}$$

$$A_2(0) + A_2' = \text{B}^{10} \text{ concentrations/reactor} = 4.47419 \times 10^{23} \text{ atoms/reactor}$$

$$A_3(0) + A_3' = \text{Li}^6 \text{ concentrations/reactor} = 7.21565 \times 10^{23} \text{ atoms/reactor} .$$