IDO-16667
AEC Research and Development Report
Reactors - General
TID-4500, Edition 15

THE ADVANCED TEST REACTOR - ATR

FINAL CONCEPTUAL DESIGN

by

D. R. deBoisblanc et al.

PHILLIPS PETROLEUM COMPANY
Atomic Energy Division

Under Contract
AT(10-1)-205

to

IDAHO OPERATIONS OFFICE
U. S. ATOMIC ENERGY COMMISSION

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November 1, 1960

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S U M M A R Y

This report presents the results of a study which was directed at providing additional experimental loop irradiation space for the AEC-DRD testing program. It was a premise that the experiments allocated to this reactor were those that could not be accommodated in the MTR, ETR or in existing commercial test reactors.

To accomplish the design objectives called for a reactor producing perturbed neutron fluxes exceeding 10^{15} thermal neutrons per square centimeter per second and 1.5×10^{15} epithermal neutrons per square centimeter per second. To accommodate the experimental samples, the reactor fuel core is four feet long in the direction of experimental loops. This is twice the length of the MTR core and a third longer than the ETR core.

The reactor concept herein proposed represents an advance in test reactor technology and can be expected to contribute a substantial body of information to reactor development as has been the case with the MTR and ETR.

The vertical arrangement of reactor and experiments permits the use of loops penetrating the top cap of the reactor vessel running straight and vertically through the reactor core. The design offers a high degree of accessibility of the exterior portions of the experiments and offers very convenient handling and discharge of experiments. Since the loops are to be integrated into the reactor design and the in-pile portions installed before reactor startup, it is felt that many of the problems encountered in MTR and ETR experience will cease to exist. Installation of the loops prior to startup will have an added advantage in that the flux variations experienced in experiments in ETR every time a new loop is installed will be absent.

The Advanced Test Reactor has a core configuration which provides essentially nine flux-trap regions in a geometry which is almost optimum for cylindrical experiments. The geometry is similar to that of a four-leaf clover with one flux trap in each leaf, one at the intersection of the leaves, and one between each pair of leaves. The nominal power level is 250 megawatts.

This study was carried out in enough detail to permit the establishment of the design parameters and to develop the power requirement which, conservatively rated, will definitely reach the flux specifications. A critical mockup of an arrangement similar to ATR was loaded into the Engineering Test Reactor Critical Facility.

The study of the remainder of the plant layout and the engineering performance of the plant incorporated the large body of test reactor experience developed at the MTR and ETR site. This design contemplates location of ATR adjacent to the northwest corner of the MTR-ETR complex. The principal features of ATR are presented below.

ATR DESIGN SUMMARY

Reactor configuration	Four-lobe
Length of reactor core, ft	4
Power level (nominal), Mw	250
Fuel region volume, liters	262
Fuel loading, kg U-235	33
Estimated operating cycle, days	>17
Power density, Mw/liter	
Maximum	2.5**
Average	0.954
Vertical power distribution, maximum-to-average	1.40
Radial power distribution, maximum-to-average	1.87**
Fuel annulus metal-to-water ratio	0.75 - 0.80
Fluxes in experiments	(*)
Primary cooling water	
Flow rate, gpm	30,000
Velocity in fuel channels, ft/sec	44
Inlet temperature, °F	130
Inlet pressure, psig	300
Reactor Δt , °F	57
Core pressure drop, psi	92

*Since the manner in which this test reactor meets the requirements of each of the intended experiments cannot be shown in this abbreviated table, the reader is referred to Sections 2.0 and 3.0.

**These values contain the 1.15 factor (See Fig. 8.4K) which allows for flux peaking near the side plates.

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1.0 INTRODUCTION

1.1 Historical

On August 5, 1959 Phillips Petroleum Company was asked to undertake the conceptual design of an advanced engineering test reactor capable of satisfying the needs of the Division of Reactor Development for additional high flux irradiation space. Phillips was asked to examine and employ the most advanced technology available, and to bring forth a design with sufficient flexibility and flux capabilities to meet the Commission's needs for some time to come. This report is a technical description of the resulting reactor designated ATR*.

Previous studies¹⁻¹² concerning advanced testing reactors were carefully reviewed before deciding the direction the design should take.** The most detailed studies were those carried out under AEC contract in 1956 and 1957. These studies included many advanced reactor types, ranging from a heavy water homogeneous system to the light-water moderated flux-trap system. All of these studies were directed at providing high fluxes in large loop facilities. This reactor is aimed at providing extremely high flux environment for a multiplicity of high-pressure loops whose greatest sample diameter is about 2 in. In this study, an attempt has been made to bring a fresh viewpoint to the problem. The cylindrical symmetry of the experimental loops, their small diameter and the large number of samples to be irradiated at one time were factors which strongly influenced the choice of the reactor type.

Operational experience in multi-facility reactors such as the MTR and ETR set an upper limit of loops at about nine, to avoid an unduly large amount of down time. In a program where many samples are being irradiated simultaneously, each with a reasonable chance of failure and in which each sample is capable of causing the reactor to be shut down, downtime becomes a dominant factor. ATR is considered a reasonable compromise.

1.2 Design Criteria

As a result of several conferences between Phillips, Idaho Operations Office and personnel of Washington AEC, the following design criteria were agreed upon following DRD review of the original conceptual design report:

1. ATR will be a four-lobe reactor substantially as described in IDO-16666, but with some modifications which are incorporated in the present report.

*This report presents the conceptual design of ATR, formally known as ETR II⁽¹³⁾ prior to that referred to as ETR IV⁽¹⁴⁾ and occasionally has been called the Advanced Engineering Test Reactor (AETR), not to be confused with the Advanced Epithermal Thorium Reactor (AETR).

**Appendix 8.2.

NOTE: References 1-14 are given at end of Section 1.0.

2. The reactor shall be capable of operating with controllable power distribution between separate lobes.

3. Control shall be accomplished in such a manner as to preserve at least a cosine vertical distribution of flux throughout each cycle of operation.

4. The reactor shall be a generalized test reactor but must meet Naval Reactors requirements as now defined.

5. The location is assumed to be northwest of the MTR-ETR complex.

In regard to Item 3, the present most promising method for achieving the required axial flux distribution plus all other requirements is the use of a chemical poison in a light-water solution, such as boric acid, in the reflector region just outside each lobe and in the neck region of each lobe. However, mechanical control systems which can accomplish the required power distribution control and axial flux distribution may be considered. Examples of the latter are systems in which the poison distribution in the vertical direction remains constant in each plane perpendicular to the axis of the reactor, as for example, poison sections in which the poison concentration varies linearly from top to bottom in two oppositely moving plates. Another example is the "variable area" type control such as used in the reflector region of the RMF and ARMF reactors at the NRTS.

In regard to reflector type, D₂O should be compared to beryllium on a realistic basis involving consideration of the engineering feasibility, ability to produce desired experimental conditions, nuclear safety and real cost. The problem of containment of D₂O must be weighed against the fast neutron damage which limits the service lifetime of beryllium.

The objectives of ATR design require some departure from test or power reactor technology in order to provide spatially uniform, time independent neutron flux fields in the greater than 10^{15} range both fast and thermal. Specifically, the need for simultaneous testing of several experimental samples in differing fast-to-thermal flux ratios and at differing absolute flux levels poses a design problem of first magnitude. Inasmuch as all of the irradiation experience obtained to date on materials in both MTR and ETR will be exceeded in something short of a year of operation of ATR, materials of construction bring up a definite design problem. Aluminum, light-water coolant, and highly enriched uranium alloyed with aluminum, which form the core materials of MTR and ETR have been operated at high flux conditions to a greater extent than any other materials. The low cost and the high state of development of production techniques for aluminum-uranium plate-type fuel elements coupled with an annual consumption of about eight hundred fuel elements make the choice of this type of element almost mandatory. With this selection, and the use of the alloy showing lowest corrosion rate at the highest temperature, reactor coolant conditions become fixed at the point of maximum heat generation in the core. Test space length and flux specifications under the best conditions of flux-augmentation (flux-trapping) determine power density and hence, heat flux which must be removed.

The use of special fuel elements having any combination of non-uniform distribution of fuel and distributed poisons affords some flexibility in design, but present almost insuperable problems for a conceptual design study. Therefore, a sound approach is to base the reactor on a valid extrapolation of present technology of fuel elements using uniform distribution of both fuel and poisons, recognizing that after ATR has been operated, just as in the case of MTR and ETR, development of special cores through the cooperative use of ATR and its critical facility will make possible savings in operating cost and will give improved operating characteristics.

In taking this course, the problem of control techniques becomes an important one. Despite the natural preference for the thoroughly proof-tested vertically moving mechanical control rods, it is recognized that most of the variations in test conditions in MTR and in ETR arise from their mechanical rods. If mechanical controls, other than safety rods, are to be used without penalty in ATR they must differ from those which have been used in virtually all existing research and test reactors and in most power producing machines, hence a mechanical development and testing program is indicated.

Power conservation techniques, necessary at these high flux levels, result in a very small core and hence offer little space in which to place control rods. The control of the reactivity by means of modulation of reflector albedo appears to fit in very well with the small core and the space available. The desire to shift the power among the lobes relative to each other requires some provision for insertion and removal of absorbers in the neck region of each lobe. A mechanical control system designed to operate in the annular region outside the fuel lobe might not fit well into the small space available in the neck region. Note that this region must be preserved for an all overriding safety control system. The chemical control system can be easily adapted to fit into both spaces available without serious interference with the safety blade system and this is perhaps one of the most compelling reasons for giving strong consideration to such a system. However, it must be demonstrated that a chemical solution can be found which is reasonably compatible with, 1) the radiation fields in the region of control, 2) the corrosion resistance of the containment vessels, and 3) the time dependent requirements of rate of change of poison concentration necessary to reliably control the reactor.

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2.0 REACTOR CORE DESIGN

In order to design a core best suited to fit the experimental requirements, it was necessary first to determine desirable and undesirable reactor core features. Concepts involving features such as homogeneous fuel and the use of plutonium were considered undesirable in view of the limited present day stage of the art. To achieve a relatively large amount of test space and efficient absorption of neutrons in experiments, the ATR designs (Figures 2.0A and 2.0B) feature a high leakage core, re-entrant core geometry, flux trapping, and a favorable geometrical arrangement of the fuel around the experiment.

With the four-lobe reactor, many of the advantages of the single annulus flux trap have been preserved and additional flux traps have been provided thus greatly increasing the available test space. Furthermore, the nuclear coupling reduces the mass of fissionable material needed for criticality below that required in four separate single annulus reactors. The philosophy of the multi-lobe reactor core geometry is applicable to essentially any number of fuel lobes. The four-lobe reactor with its nine excellent experimental locations was selected because it provides the optimum ratio of good experimental positions per fuel lobe and a reasonable number of loop facilities.

2.1 Fuel Disposition

Fuel elements whose cross sections are segments of a circular ring are arranged to form four partially complete right cylindrical shells, or annuli of fuel, so oriented to one another that they may be connected together by four other partially complete annuli of fuel thus forming a continuous serpentine fuel configuration.

2.2 Irradiation Spaces

The light-water filled space within the fuel configuration provides five very attractive flux traps-- one in the center of each of the four lobes and one in the center of the total fuel configuration as defined by the convex surfaces of the connecting fuel elements. The concave surfaces of these same connecting elements provide four additional valuable test spaces. The desired flux spectrum in each location is achieved by varying the metal-water ratio (by means of aluminum filler pieces) in the flux-trapping region between the fuel and the experiment.

2.3 Reflector Arrangement

Reflectors of D_2O or beryllium (and possibly zirconium- H_2O , aluminum- H_2O mixtures) should be considered as competitive possibilities in the detailed design. The reflector region commences just outside the lobe control region and extends out the necessary distance to give the desired fraction of the infinite albedo. For D_2O this amounts to about 60 cm and extends out to the wall of the main pressure vessel. For beryllium the thickness ranges from 15 cm outside each lobe to about 25 cm outside each exterior test position.

2.4 Reactor Control

Safety rods are provided in the neck region of each lobe. These are normally fully withdrawn during operation and can override all other control elements to shut down the reactor.

Shim control regions are provided just outside each lobe and in the neck region of each lobe. Figure 2.0A shows the chemical control arrangement. Figure 2.0B shows schematically mechanical control blades outside the lobes, it being understood that mechanical shims would also be required in the neck regions. Although the chemical control system is associated with a D_2O reflector (Figure 2.0A) and the mechanical system with beryllium (Figure 2.0B) these arrangements may be exchanged providing thereby four design possibilities.

FIG. 2.0A
CORE CROSS SECTION
ATR FOUR-LOBE REACTOR

P.P. Co. - C-3012

3.0 REACTOR PHYSICS

3.1 Scope and Objectives

The requirements and specifications that are dependent upon the reactor physics of the system have been determined for the conceptual design of ATR but no attempt has been made to achieve a final optimization. The material presented in this section is based upon a series of calculations which is described in detail in a report now being prepared for publication and in a previously issued report.¹ The calculations have fixed the reactor physics requirements to the extent necessary to assure that, if they are met, satisfactory performance of ATR can be attained.

The feasibility of meeting the requirements has been established for a reactor with a D₂O reflector and chemical neck and reflector shim controls which vary the boric acid content of light water. This system has been selected as the reference reactor for the conceptual design, because from the reactor physics standpoint it offers a straightforward method of holding flux distortions and variations to the required limits. Alternate reflectors, such as beryllium, and control systems, such as mechanical shims, may be considered provided that they preserve the features of the reference reactor and meet the design criteria of Section 1.2.

An extensive set of calculations has been made for D₂O reflected reactors with chemical shim control. Alternate reflectors and control systems have been evaluated by calculating the differences from the reference reactor using one-dimensional parametric studies and normalizing with two-dimensional calculations. It is assumed that the test fluxes for a specified experimental loading is determined by the power distribution in the core and that the behavior during operation with different control systems is dependent only on the ability to control this power distribution, if the requirements on vertical distortion are met.

To determine the reactor physics requirements and to investigate the reference reactor and alternate systems, the following general types of studies have been carried out:

1. Determination of the flux and power distributions throughout an operating cycle.
2. Establishment of the fuel loading, cycle time, and the control requirements and feasibility.
3. Investigation of xenon instability, coefficients of reactivity and neutron lifetime.

1. D. R. deBoisblanc et al., "Proposal for an Advanced Engineering Test Reactor - ETR II", Phillips Petroleum Company, IDO-16666, March 1960.

The calculations of initial flux and power distributions and reactivity effects have been made by two-group, multiregion, diffusion theory using the IBM-704 PDQ-2 and PDQ-3 codes. Changes throughout an operating cycle including burnup effects have been calculated with the TURBO program. Constants have been obtained from MUFT IV and modified DONATE routines. The results of the diffusion theory calculations have been compared with measurements on an experimental mockup and with higher order approximations to transport theory. These comparisons have justified use of the diffusional calculational methods for this conceptual design.

Previous studies on high flux reactors by Phillips Petroleum Company and others have been used to restrict the range of variables considered.^{1,2,3} The work_{4,5,6} of Internuclear Company has been particularly valuable for this purpose.

3.2 Design Requirements and Specifications

ATR is designed to:

1. Operate safely under all conditions including any credible failure in the experimental facilities or shim control system.
2. Provide high fast and thermal neutron fluxes in nine loop type experimental facilities.
3. Permit a considerable degree of control of the flux level in an experiment independently of the levels in other experiments.

-
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4. Minimize variations in the test fluxes during operation from burnup of fuel and manipulation of reactivity controls.

5. Minimize vertical distortions of the test fluxes.

6. Utilize reflector neutrons to improve the transient behavior.

In order to accomplish these objectives a number of requirements dependent upon reactor physics are imposed upon the design. These requirements are discussed individually and summarized in Table 3.2A.

A generalized plan of the reactor core, reflector, and control arrangement is shown in Figure 3.2A. This figure is not intended to be dimensionally correct or accurate in detail. In particular, the double barrier between the H_2O reflector shim and the D_2O reflector of the reference reactor is not shown. A more complete description of the reactor core design is given in Section 4.0 of this report. The north, east, south, and west fuel lobes or tests as a group are designated the outer lobes or tests. The southeast, northwest, etc., tests are referred to as the exterior tests.

3.21 Reactor Core

The core specifications important to the reactor physics are listed in Table 3.2B.

TABLE 3.2B

REACTOR CORE SPECIFICATIONS

Active Height	48 in.
Thickness	2.5 in.
Volume	262 liters
Metal-to-Water Ratio (Volume)	0.8
Cladding and Structural Material	Aluminum
Fuel Alloy	Highly Enriched U-235 in Aluminum
Fuel Loading	Approximately 33 kg U-235

The height of 48 inches is selected as the minimum which will hold the axial flux variation to the desired limits. Any substantial increase in the height will introduce serious problems from axial xenon oscillations as discussed in Section 3.7.

TABLE 3.2A

ATR SPECIFICATIONS

Core Height	48 in.
Core Thickness	2.5 in.
Core Volume	262 liters
Core Metal-to-Water Ratio (Volume)	0.8
Core Structural Material and Cladding	Aluminum
Fuel	93% Enriched U-235
Fuel Loading	33 kg U-235
A-5 Experiment Fast Flux - Avg Midplane Value	1.0×10^{15} n/cm ² -sec
A-5 Experiment Thermal Flux - Avg Midplane Value	0.32×10^{15} n/cm ² -sec
A-3 Experiment Fast Flux - Avg Midplane Value	1.5×10^{15} n/cm ² -sec
A-3 Experiment Thermal Flux - Avg Midplane Value	0.48×10^{15} n/cm ² -sec
A-1 Experiment Fast Flux - Avg Midplane Value	1.0×10^{15} n/cm ² -sec
A-1 Experiment Thermal Flux - Avg Midplane Value	1.0×10^{15} n/cm ² -sec
Control Requirements	
Safety Blades	10.0% $\Delta k/k$
Neck Shims	2.9% $\Delta k/k$
Reflector Shims	6.5% $\Delta k/k$
Burnable Poison	4.8% $\Delta k/k$
Core Power	250 Mw
Maximum Power Density	2.5 Mw/liter
Vertical Maximum-to-Average Power Ratio	1.4 or less
Cycle Time at Full Power	17 days

The thickness of 2.5 inches represents a compromise. A thinner core reduces the required power for given test fluxes but increases the maximum power density and a thicker core produces the opposite effects. Final optimization may indicate the desirability of a minor change in the core thickness.

The core shape can only be specified by a detailed drawing but, of course, the values and control of the test fluxes are dependent upon the cloverleaf shape of Figure 3.2A with its several isolated flux traps. The core volume being fixed by the height, thickness, and shape need not be considered as a primary specification.

The core metal-to-water volume ratio of 0.8 is set by both nuclear and heat removal considerations. Small variations do not have a large effect on the reactor performance and this value may change slightly in the final design.

The selection of aluminum as the structural material is based upon both nuclear and non-nuclear factors. The use of highly enriched, greater than 93 atom per cent U-235, uranium is essential to the design of this high flux reactor. The use of an aluminum-uranium metal fuel alloy keeps the absorption cross section low while permitting use of a well developed method of construction. Any substantial modification of the materials recommended will require a complete review of the reactor physics of the design.

The fuel loading of 33 kg is to be considered as only an approximate value. The actual requirement will be dependent upon design details and the final decision as to the length of the operating cycle. However, the amount of change required will not be such as to introduce problems in construction of fuel elements, control of reactivity, or attainment of a satisfactory amount of fuel burnup.

3.22 Fluxes in Experiments

The flux requirements to be met or exceeded are given in Table 3.2C along with the available fluxes at a reactor power of 250 megawatts. The experiments all have a diameter of 2 inches and a uniform composition with a metal-to-water volume ratio of one and a U-235 content which produces a fission heat of 200 kw per ft at the thermal flux level specified in Table 3.2C. This table gives values averaged over the midplane of a test in a stainless steel facility, the dimensions of which are shown in Figure 3.3A. The designations A-1, A-3, and A-5 of the tests are determined by the flux requirements as shown in Table 3.2C. The test fluxes are dependent upon the power distribution in the core and the compositions of the flux trap moderators; thus, the available fluxes in Table 3.2C apply only to a typical power distribution and experimental loading. In addition to meeting or exceeding the individual fast and thermal flux requirements in a test, it is desirable to hold their ratio close to the requirements of Table 3.2C.

TABLE 3.2C

ATR FLUX REQUIREMENTS AND PERFORMANCE

AVERAGE MIDPLANE FLUXES IN TESTS

Test Type	Test Position	Fast Flux $\text{n/cm}^2\text{-sec} \times 10^{-15}$		Thermal Flux $\text{n/cm}^2\text{-sec} \times 10^{-15}$		Fast to Thermal Flux Ratio	
		Required	Performance at 250 Mw	Required	Performance at 250 Mw	Required	Performance
A-3	Center	1.5	2.1	0.48	0.61	3.1	3.5
A-5	Outer	1.0	1.3	0.32	0.44	3.1	3.0
A-5	Exterior	1.0	1.6	0.32	0.56	3.1	2.9
A-1*	Outer	1.0	1.3	1.0	0.66	1.0	2.0

* The thermal flux requirement for the A-1 experiment can be met by use of an inpile tube with nuclear properties equivalent to zirconium or with a stainless steel tube by placing it in the center and using non-uniform fuel distribution to reduce the peak power density.

The average fast flux in any experiment should not vary more than plus or minus 10 percent from the mean value during an operating cycle. If this requirement is met, the variation of the thermal flux will be satisfactory. The amount of fission heat from a test changes because of burnup of fuel in the test and the allowable variation of the fission heat cannot be specified. The axial maximum-to-average ratio of the thermal flux in a uniform test specimen should not exceed 1.4 at any time during the cycle.

3.23 Reactor Controls

The reactor control system must compensate for loss of reactivity from burnup of fuel, burnable poison, and formation of fission products, provide adequate shutdown capability at all times, and maintain the required power distribution among the core lobes. The reactivity controls are divided into two major types: The safety controls and the operating controls.

The safety controls utilize blades which when inserted introduce poison into the core. They are always completely withdrawn when the reactor is operating. These controls must be fast acting to provide protection against reactivity transients. A minimum worth of 10 percent $\Delta k/k$ with the neck shim controls unpoisoned is required. This requirement assures adequate shutdown capability even if all boron is lost from the shim controls. To prevent one of the rather loosely coupled lobes going critical by itself, the requirement is imposed that the reactor must be subcritical with any one shim blade withdrawn and the others inserted.

The operating controls consist of the reflector and neck shim controls and burnable poison in the core. The shim controls are used to adjust the initial power distribution among the core lobes and to maintain it during operation. This adjustment of the power distribution is the means by which the desired initial flux levels in the test are achieved and maintained through a cycle. Therefore, there are different requirements on each neck and reflector shim control and, thus, individual control of each neck and reflector shim is required. The operating controls must not produce flux distortions which cause the axial component of the maximum-to-average power ratio in the core to be greater than 1.4 at any time during the cycle. The control requirements are summarized in Table 3.2D. An allowance is included for the effects of non-uniform burnup of fuel in the axial direction. These requirements are based on a seventeen day cycle in which test fluxes do not vary more than plus or minus 10 percent.

TABLE 3.2D

ATR REACTIVITY CONTROL REQUIREMENTS

<u>Control</u>	<u>$\Delta k/k$ - percent</u>
Safety Blades	10.0
Four Neck Shims	2.9
Four Reflector Shims	6.5
Burnable Poison in Core	4.8

3.24 Power and Power Density

The core must be capable of a power production of 250 megawatts. The limit on the power is set by the surface temperature at the point of maximum power density. The heat transfer conditions on which the recommended maximum of 2.5 megawatts per liter is based are discussed in Section 4.5 and Appendix 8.7. By changes such as increased coolant velocity or acceptance of nucleate boiling, it may be possible to permit a substantially higher maximum power density. Calculations indicate that the maximum power density in ATR might be as high as 2.8 megawatts per liter for an extreme combination of core power shifting and test thermal flux requirement. Every effort should be made in the design to prevent unnecessary flux peaking. The maximum power density should be determined by flux measurements on a critical mockup, and core power and heat removal conditions made compatible with this value.

3.3 Calculational Methods

The majority of the calculations of reactivity and flux distributions for this study have used two-group diffusion theory. Four-group diffusion theory calculations and higher order approximations to transport theory have been used to check the validity of the two-group calculations. A consistent method has been used for obtaining the diffusion theory constants for all calculations. The programs and methods used in obtaining the constants are well known and therefore the constants are not tabulated in this report.

The thermal constants have been averaged over a Maxwellian neutron energy spectrum by an IBM-650 program. The cross section data for the thermal group are essentially the same as those of the SUFFOCATE program. The constants have been averaged over a Maxwellian spectrum rather than the Wigner-Wilkins spectrum because while this procedure produces a somewhat larger error in the calculated value of k_{eff} it reproduces more closely the flux depressions in regions of heavy absorption.

The fast constants have been calculated by the MUFT IV program for the IBM-704 computer. For the nonhydrogenous moderators D_2O and beryllium the MUFT IV program is preferable to the MUFT III program for the IBM-650 because MUFT IV permits the use of age numbers obtained by the method of Greuling-Goertzel. The constants as obtained by these methods have been found to give reasonably good results in criticality calculations.

The two-dimensional calculations of the ATR in x-y geometry have been made with the PDQ-3 program for the IBM-704. It is assumed that the addition of 30 cm reflector savings in axial direction reduces the problem to that for a bare reactor in the axial direction and together with the assumption of separability of axial and radial flux distributions reduces the three-dimensional problem to one that can be calculated in two dimensions. The effects of axial flux distortions have been calculated for a single lobe model in r-z geometry using the PDQ-2 program.

The one-dimensional calculations have used the WANDA code on the IBM-704 and the DMM code on the IBM-650. These codes solve the one-dimensional few-group diffusion equations.

A number of burnout studies have been made using CANDLE which is a one-dimensional few-group burnout code. Several studies were made using the two-dimensional burnout code (TURBO) with the same mesh as used for PDQ calculations.

Other codes used to study local thermal flux variations were the S_n and double P_1 transport codes. The results from these higher order approximations to transport theory have been compared with the diffusion theory calculations to justify their use for ATR with its many thin regions differing widely in absorbing and scattering properties.

A number of different PDQ mesh descriptions of ATR have been used. The flux distributions have been determined using quarter core descriptions with almost the maximum number of mesh points, 7,500, allowed by PDQ-3. Effects of controls and perturbations on the flux distributions have been determined with quarter or half core descriptions using a coarse mesh to reduce the machine time for the calculations. The results of the coarse mesh problems have been normalized by comparison with reference to fine mesh calculations.

A quarter section of a typical model and its representation in rectangular geometry for two-dimensional PDQ studies of ATR is shown in Fig. 3.3A. This model is not to be considered as an accurate dimensional representation of the conceptual design. In particular region 2, the reflector shim control or inner reflector is shown as a three-inch thick region. The reference conceptual design has a light water annulus with variable poison content and retaining tank walls in this region. If an alternate control system is selected, this region could be considered as representing the control and its supports and possibly part of the other reflector.

Region 1 shows the outer reflector which is clean D_2O in the reference conceptual design, but might represent a solid reflector in an alternate design. Region 3 is the reflector for the exterior experiment. This region is part of the D_2O outer reflector in the conceptual design but is represented separately to permit study of alternate reflectors. Regions 4 and 11 constitute the homogenized active core. In the complete model the core is represented by a number of regions to facilitate study of power shifting among the core lobes.

Region 5 represents the tanks used to separate regions of different coolant composition and flow conditions. It is included to make the calculational model more realistic but is not to be considered as an accurate representation of the mechanical design. Regions 6, 12, and 17 are the flux trap moderators. Their composition depends upon the desired fast-to-thermal flux ratios. For an A-3 test in the center position the composition of region 12 should be about 30 per cent light

water and 70 per cent aluminum. For A-5 tests in the outer and exterior positions the composition should be about 40 per cent aluminum and 60 per cent water. An A-1 test requires a flux-trap moderator consisting only of light water.

Region 7 represents the loop pressure tubes which are stainless steel and the gas annulus. Regions 8, 9, and 10 are the test regions. The metal-water ratio of these tests is 1.0 and the uranium content is set by the test specifications. The test regions are considered to have an effective neutron temperature of 400°F. All other regions are assumed to possess an effective neutron temperature of 200°F. Most of the neutrons captured in a test suffer their last scattering collision in the flux-trap moderator which has a 200°F temperature. With allowance for some spectral hardening in the surrounding stainless steel tubes the 400°F effective neutron temperature is considered to give absorption cross sections which yield a satisfactory value for the fission heat when the loop coolant temperature is 650°F.

Region 13 is the neck filler and has a composition of 20 per cent water and 80 per cent aluminum in the conceptual design. Regions 14 and 15 represent the neck shim control and consist of light water with variable poison content in the reference design. In an alternate design these regions might represent a mechanical shim control. Region 16 represents a safety blade which is black to thermal neutrons when inserted and consists of 20 per cent light water and 80 per cent aluminum when withdrawn. In the reference design these blades are extended in arcs along the outside edge of the core to increase their shutdown worth.

3.4 Flux and Power Distributions

The fluxes in the tests at a reactor power of 250 megawatts have been presented in Table 3.2C. These values, except as otherwise stated, are for a core power distribution in which each of the outer core lobes has approximately 19 per cent of the total power and the center fuel region 23 per cent. The flux levels in the A-1 experiment are for a power distribution with 22 per cent of the total core power in the lobe containing the test with a flux trap composition consisting entirely of light water.

It is seen that at 250 megawatts the specified flux levels are substantially exceeded for the A-3 and A-5 tests. The flux levels in the exterior positions are quite satisfactory. However, it should be recognized that these flux levels cannot be controlled independently of the levels in the outer lobe tests.

The variations of the fast fluxes in the tests with reactor operation are shown in Fig. 3.4A and those of the thermal fluxes in Fig. 3.4B. These flux changes have been computed using the TURBO burnup program and neglect changes caused by non-uniform burnup of fuel in the axial direction. It is seen that for 17 days of 250 megawatt operation the control of the test fluxes is excellent. The departures after that time are the result of loss of ability to control the core power distribution as discussed in

Section 3.5. Burnup of fuel in the tests is shown in Fig. 3.4C. The burnup of fuel in the core and tests causes the fast-to-thermal flux ratio to decrease in a linear manner by about 15 per cent during a 17 day cycle.

The maximum power density in the core is dependent upon the core power distribution as well as the factors previously discussed. The calculated maximum power density of 2.8 megawatts per liter at a core power of 250 Mw is for a flux-trap moderator of pure water and a power distribution in which two of the outer lobes each generate 16.5 per cent of the total power, the other pair of outer lobes each 22 per cent and the center lobe 23 per cent, and for an axial maximum-to-average power ratio of 1.4. Since one pair of outer lobes is operating at a power of 1.3 times that of the other pair and the water adjacent to the core increases flux peaking, the maximum power density is given for a rather extreme situation. The value of 2.8 megawatts per liter is based upon maximum-to-average ratios that include the epithermal fissions and is adjusted for the rectangular representation of the reactor in PDQ and TURBO. The behavior of the maximum power density during operation, as calculated by TURBO, is shown in Fig. 3.4D. It is seen that over the period of the 17-day cycle the maximum power density in the core continually decreases.

3.5 Control of Reactivity

3.51 Cycle Time and Control Requirements

The control system of ATR must not only compensate for changes in reactivity from fuel burnup, fission product formation and consumption of burnable poison; but must also maintain the proper power distribution among the core lobes through the cycle. This requirement of correct power distribution increases the required worth of the shim controls both individually and as a group. The control through a cycle of the reference reactor using chemical reflector and neck shim controls has been investigated using TURBO.

The control of the individual lobe powers during operation is shown in Figure 3.5A. It is seen from this figure that the length of time that ATR can operate at full power before refueling, the cycle time, is limited by the ability to hold the required core power and test flux distribution, rather than by loss of criticality. The loss of control over the power distribution after about 17 days, which sets the limit on the cycle time, occurs because all of the poison has been removed from the necks. If the change in power distribution and test fluxes, Figures 3.4A and 3.4B, after 17 days is permissible then the reactor can operate at full power for about 31 days before it is necessary to refuel. It is shown in Figure 3.4D that no additional problem from the maximum power density results during this longer cycle. The burnup of fuel in the core during a cycle is shown in Figure 3.5B. After 17 days of full power operation the average burnup of fuel in the core is 20 per cent and after 31 days it is 36 per cent. The variation with operating time of k_{eff} with neck and reflector shim controls removed but with burnable poison retained in the core is shown in Figure 3.5C.

3.52 Control Systems and Reflectors

The worths of the ATR controls have been determined by calculations of the excess reactivities of two-dimensional models using the PDQ code. Results are given for chemical neck and reflector shim controls, uniformly distributed natural boron in the core, and for blade type safety controls. The requirements which this or an alternate control system must meet have been presented in Section 3.2.

The calculated worths of the safety controls are for blades extending across the necks to the outside edges of the fuel regions as shown in Figures 3.2A and 3.3A. The worth of these blade structures is 5.1 per cent $\Delta k/k$ when the neck shims are clean and 3.0 per cent when the necks are poisoned. These values do not meet the requirement of 10.0 per cent. Therefore, in the conceptual design the blade structure is extended in an arc around the outside edge of the core as shown in Figure 2.0A so that adequate worth is available at all times. The safety blades are always completely withdrawn when the reactor is operating. Care should be taken in the design of the follower section and guide structure to prevent excessive flux peaking in adjacent fuel regions.

The worth of a reflector shim control system consisting of four 1/2 in. thick water annuli extending about 180 degrees around each core lobe is 5.6 per cent from the unpoisoned condition to a content of 50 g of boric acid, containing natural boron, per liter. By increasing the arc around each lobe to 230 degrees, as is geometrically feasible, the worth is increased to an estimated value of 7 per cent $\Delta k/k$. Increasing the thickness from 1/2 in. to 3/4 in. without lengthening the arc increases the worth to about 6.2 per cent. Thus, there is no difficulty in meeting the required worth of 6.5 per cent.

The neck shim controls have a worth of 4.3 per cent from clean water to 50 g of boric acid per liter. The neck shims are used largely to produce an outward power shift counteracting the effect of the reflector so that control of the power distribution can be maintained during the cycle. Thus, the location should be chosen to optimize the ability to produce an outward power shift. As only 2.9 per cent worth is required the neck controls are adequate providing the reflector shim produces sufficient inward power shift. This condition makes a sufficient amount of the neck control worth available for power shifting.

The outer reflector influences the required fuel loading by its efficiency in returning neutrons to the core. It also has an important effect upon the neutron lifetime. D_2O is selected for the outer reflector of the ATR conceptual design because of the long neutron lifetime provided by this material. Changing from D_2O to a beryllium reflector with a minimum thickness of 6 in. and with 10 volume per cent of light water for cooling does not change the required fuel loading of 33 kg of U-235. The lower absorption cross section of D_2O is offset by the extra neutron loss in the retaining vessels as far as the critical mass is concerned.

For clean reflector shim control beryllium gives an average lifetime for the neutrons from the reflector of about 400 μsec as compared to about 1000 μsec with D_2O .

3.53 Control of Perturbations and Power Distributions in the ATR Core

The loose coupling between lobes in ATR introduces the property that a perturbation in the operating conditions of one lobe can cause a shift of power from lobe to lobe. The control system must be capable of holding deviations from the desired lobe power levels within acceptable limits. In addition, as previously discussed, the control system must be capable of producing the initial power unbalanced among the core lobes to give the desired test fluxes. The effects on the core power distribution of changes in the absorption cross section of individual core lobes and of the controls individually and in groups are listed in Table 3.5A.

TABLE 3.5A

EFFECTS OF CONTROLS AND CORE POISONING ON ATR POWER DISTRIBUTION

	k_{eff}	Power-Megawatts				% Shift + increase - decrease			
Core Lobe		E	W	N & S	C	E	W	N & S	C
Half Core Cases									
Reference Case $\Sigma^p = .015$ all lobes	1.224	50.6	49.9	50.2	49.1	---	---	---	---
$\Sigma^p = 0$ in east lobe	1.254	75.6	38.1	45	46.4	+49.4	-23.6	-10.4	-5.5
$\Sigma^p = .035$ in east lobe	1.207	33.2	58.4	53.9	50.7	-34.4	+17.0	+ 7.4	+ 3.3
3 G/L boric acid in east reflector shim	1.208	32.5	58.3	53.5	52.3	-35.8	+16.8	+ 6.6	+ 6.5
10 G/L boric acid in east reflector shim	1.202	24.9	62.6	54.6	53.4	-50.8	+25.5	+ 8.8	+ 8.8
22 G/L boric acid in east reflector shim	1.200	22.4	64.1	54.9	53.7	-55.7	+28.5	+ 9.4	+ 9.4
10 G/L boric acid in east neck shim	1.220	49.5	50.5	50.8	48.5	- 2.2	+ 1.2	+ 1.2	- 1.2
50 G/L boric acid in east neck shim	1.213	46.1	53.1	52	46.9	- 8.9	+ 6.4	+ 3.6	- 4.5
Quarter Core Cases			Same as E	Same as E			Same as E	Same as E	
Reference Case $\Sigma^p = .015$ all lobes	1.225	50.2			49.2	---	---	---	---
$\Sigma^p = 0$ in center lobe	1.252	48.2			57.2	- 4.0			+16.3
$\Sigma^p = .035$ in center lobe	1.200	52.1			41.5	- 3.8			+15.7
3 G/L boric acid in all reflector shims	1.151	45.8			66.9	- 8.8			+36.0
10 G/L boric acid in all reflector shims	1.119	43.1			77.7	-14.1			+57.9
22 G/L boric acid in all reflector shims	1.108	41.9			82.3	-16.5			+67.3
10 G/L boric acid in all neck shims	1.208	51.0			46.2	+ 1.6			- 6.1
50 G/L boric acid in all neck shims	1.180	52.6			39.8	+ 4.8			-19.1
200 G/L boric acid in all neck shims	1.169	53.5			35.9	+ 6.6			-27.0

G/L boric acid = grams of boric acid containing natural boron per liter

The addition of 0.02 cm^{-1} poison in the east lobe fueled region causes a power reduction in the east lobe of approximately 34 per cent. In order to restore this power to its value before the addition of the poison it is necessary to pull poison out of the east reflector. If originally there were 3 g/liter of boric acid in the east reflector and this is pulled out, the power in the east fueled region will be restored to or slightly above what it was before the introduction of the 0.02 cm^{-1} poison in the fueled region.

Poison in a single neck region has a minor effect on power shifting when compared to the poisoning in a reflector region.

Conclusions are that the combination of reflector and neck poison certainly gives adequate control for power restoration from disturbances while operating.

Under more realistic operating conditions poison will be found in all necks and reflectors. As can be seen a great deal of power shifting can be accomplished by a combination of removing poison from one reflector and adding it to another.

The ability of the control system to maintain the required power distribution after shutdown from full power has been investigated by a TURBO calculation. After 200 hours of operation at 250 Mw the reactor was shut down to 2.5 Mw for one-half hour and then returned to full power.

The difference in the power densities of the lobes causes the loss of reactivity from nonequilibrium xenon formation after shutdown and subsequent return to full power to tend to decrease the power in the lobe where the highest power is desired (the center lobe) and to increase it where the lower powers are wanted. Thus after return to full power it is necessary to exert all available shim control in the reflector in order to shift the power as far as possible from the outer lobes into the center lobe. Then, as the burnup of the nonequilibrium xenon proceeds, more control becomes available and it again becomes possible to obtain the desired power balance between lobes by use of all control regions. The desired power balance is obtained within one-half hour after return to full power. Immediately after startup the worst power deviation occurs with the center lobe power being 14 per cent lower than desired and each of the outer lobe powers being 4 per cent higher than desired. After burnup of the nonequilibrium xenon is complete, the xenon concentration will be at a lower value than when the reactor was shut down and more excess reactivity is available than at shutdown. The effects on reactivity, power, and fluxes are shown in Figures 3.5D, 3.5E, 3.5F, and 3.5G.

3.6 Void and Temperature Coefficients

The temperature coefficient of reactivity is determined by the net balance of a number of positive and negative effects but in ATR is primarily due to changes in the density of the coolant and moderator fluids in the reactor, i.e., by the void coefficient.

Moreover, the possibility of boiling at various locations and the introduction of cooler and more dense water into the different regions of the reactor place additional emphasis on a knowledge of the void coefficient of reactivity. The variation of the void coefficient of reactivity with position (mainly radial position) allows the further possibility that the reactivity change can be different in sign as well as magnitude if boiling in a region takes place uniformly rather than near some hot spot near the surface of the region.

The void coefficient as a function of radial position in a single lobe model of ATR has been determined with a series of one-dimensional diffusion theory studies performed using the WANDA-4 code. The studies were based on the simplifying assumption that initial study of void coefficients in a single detached lobe can be directly related to those of the complete reactor by the effect of reactivity change in one lobe upon net reactivity of the complete reactor. The reactivity changes for the single lobe model as listed in Tables 3.6A and 3.6B are about three times the corresponding values for ATR.

All void coefficients were calculated from the difference in the k value (calculated by the WANDA-4 code) of the "reference" reactor and the reactor with a small finite change in the H_2O and D_2O density in some radial annulus. In the test region, the filler, and the core, the change in reactivity (k) was observed for a change in the density throughout the region, in only the inner $1/4$ in. of the region, and in only the outer $1/4$ in. of the region. A complete list of all of the cases run is given in Tables 3.6A and 3.6B, together with the Δk values which resulted. Also listed in this table are the values $\Delta k/\Delta V$ and $\Delta k/v$ for each case.

The void coefficient of reactivity is assumed to be simply $\Delta k/\Delta V$, where Δk is the observed difference in k equal to $k - k_0$ (k_0 is the k of the reference case), and ΔV is the effective void which results from the change in H_2O or D_2O density. For example, if the Δk results from changing the H_2O density in a region from ρ_0 to $0.75 \rho_0$, the ΔV is simply 0.25. Hence, ΔV is the fraction of the H_2O which is replaced by void.

In order to obtain a meaningful measure of the radial position dependence of the void coefficient, a set of numbers termed $\Delta k/v$ was also computed. These numbers represent the change in k produced by adding one cm³ of void per cm of reactor length at each of the respective positions. Hence $\Delta k/v = \Delta k/(\Delta V \times \text{Volume of region perturbed})$.

TABLE 3.6A

VOID COEFFICIENT RESULTS FOR A-1 TEST WITH 8 GMS PER LITER OF H_3BO_3 IN CONTROL REGION

Case No.	Perturbed Region	Perturbation, ΔV = Void Fraction of H_2O	Volume of Region (cm^3/cm)	K	$\Delta K \times 10^2$	$\frac{\Delta K}{\Delta V} \times 10^2$	$\frac{\Delta K}{V} \times 10^4$
0	Reference Case			.91573148			
1	Inner 1/4" of Test	0.25	1.2668	.91578024	.004876	.019504	1.5397
2	Entire Test	0.25	20.2683	.91653108	.079960	.31984	1.5780
3	Outer 1/4" of Test	0.25	8.8674	.9160965	.036502	.14601	1.6466
4	Inner 1/4" of Filler	0.10	14.2527	.9165406	.080907	.80907	56.766
5	Entire Filler	0.10	105.46	.92034215	.461067	4.61067	43.720
6	Outer 1/4" of Filler	0.10	26.6021	.91667063	.093915	.93915	35.304
7	Inner 1/4" of Core	0.10	31.6692	.91519423	-.053724	-.53724	-1.6964
8	Entire Core	0.10	430.7012	.89145217	-.2427931	-.242793	-5.6372
9	Outer 1/4" of Core	0.10	54.4710	.91432276	-.140872	-1.40872	-2.5862
10	Inner 1/4" of Control Region	0.10	59.5381	.91523149	-.05000	-.5000	-.83978
11	Entire Control Region	0.10	881.6705	.91597369	.024221	.24221	.02747
12	D ₂ O Reflector	0.10	31,693.	.91218641	-.354507	-3.54507	-.01119
13	Control Region and D ₂ O Reflector	0.10	32,575.	.91164447	-.408701	-4.08701	-.01255
14	Entire Test	1.00	20.2683	.9179775	.22460	.22460	1.1081
15	Middle 2.5" of Core	0.10	344.5610	.89354987	-2.218161	-22.18161	-6.4376
16*	Reference Case			1.2201704			
17	Middle 2.5" of Core	0.10	344.561	1.2091922	-1.09782	-10.9782	-3.1861
18	Entire Core	0.10	430.701	1.2104614	-.9709	-9.709	-2.2542
19	Inner 1/4" of Control Region	0.10	59.5381	1.2193461	-.08242	-.8243	-1.3845
20	Entire Control Region	0.10	881.6705	1.2106277	-.95427	-9.5427	-1.0823
21	D ₂ O Reflector	0.10	31,693.	1.1981379	-2.20325	-22.0325	-.069518
22	Control Region and D ₂ O Reflector	0.10	32,575.	1.1877533	-3.24171	-32.4171	-.099515

* In Cases 16 through 22 there is no H_3BO_3 in the control region.

TABLE 3.6B

VOID COEFFICIENT RESULTS FOR A-5 TEST WITH 8 GMS PER LITER OF H_2O IN CONTROL REGION

Case No.	Perturbed Region	Perturbation, $\Delta V =$ Void Fraction of H_2O	Volume of Region (cm^3/cm)	K	$\Delta K \times 10^2$	$\frac{\Delta K}{\Delta V} \times 10^2$	$\frac{\Delta K}{V} \times 10^4$
23	Reference Case			.8732584			
24	Inner 1/4" of Test	0.25	1.26677	.8732615	.00031	.00124	.098
25	Entire Test	0.25	20.2683	.8734875	.02291	.09162	.4520
26	Outer 1/4" of Test	0.25	8.8675	.8734006	.014217	.05687	.6413
27	Inner 1/4" of Filler	0.3575	25.6779	.8732750	.001655	.004629	.01803
28	Entire Filler	0.3575	116.8853	.8729031	-.035537	-.09940	-.08504
29	Outer 1/4" of Filler	0.3575	26.6021	.8731346	-.012384	-.034641	-.1302
30	Inner 1/4" of Core	0.10	31.6692	.8718496	-.14088	-1.4088	-4.4486
31	Entire Core	0.10	430.7012	.8443828	-2.88756	-28.8756	-6.7043
32	Outer 1/4" of Core	0.10	54.4710	.8716190	-.16395	-1.6395	-3.0098
33	Inner 1/4" of Control Region	0.10	59.5381	.8727535	-.05050	-.5050	-.84813
34	Entire Control Region	0.10	881.6705	.8738043	.05459	.5459	.06191
35	Entire Test	1.0	20.2683	.8738670	.060857	.060857	.30026

Whether the lobe contains the A-1 or the A-5 test, the positive void coefficient in the test is reassuringly small and relatively constant with radial position. Cases 14 and 35, respectively, refer to the entirely dry test region for the A-1 and A-5; and in both of these cases the Δk is less than that needed for prompt neutron reactivity. The A-1 test has the higher void coefficient by about a factor of 3.5 as compared to the A-5 test, but when even the A-1 is dry the reactivity increase is more than a factor of 3.0 away from that needed for prompt criticality.

In the 40 per cent H_2O - 60 per cent aluminum filler which is used with the A-5 test, the void coefficient is small and it is positive only at the extreme inside boundary (near the test). The void coefficient for voids distributed uniformly is negative and it apparently becomes more and more negative toward the core. However, in the 100 per cent H_2O filler used with the A-1 test the void coefficient is rather large and positive throughout the region with the highest value at the boundary near the test.

The indication is that a uniform distribution of about 16 per cent void in the A-1 filler region represents enough Δk to meet the usual criterion for prompt criticality (about .72% $\Delta k/k$) in a single lobe model. There is considerable reassurance, however, in the fact that such a reactivity rise would have to occur relatively rapidly and in all lobes simultaneously in order to be hazardous and considerable bulk boiling would have to occur in order to attain this void fraction. In addition, the possibility of nucleate boiling in the region adjacent to the core or the test does not appear to represent a hazard even should it occur, for the results indicate that about a 100 per cent void fraction is necessary in the 1/4 in. layer near the core or near the test in all lobes at the same time in order to meet the usual condition for prompt criticality.

Two kinds of dependence are indicated for the void coefficient in the core. First, the coefficient becomes much more negative in the center of the core annulus than at the edges; and, second, the coefficient becomes more negative as poison is added to the control region outside the core. The void coefficient for uniform distribution of voids in the core is over twice as large as the value near the edges, and the average core void coefficient with 8 g/liter of H_3BO_3 in the control region is about twice as large as the value obtained with the control region clean.

In the control region the void coefficient near the core is negative whether or not the reflector control is poisoned and it is only slightly more negative when it is unpoisoned. In the unpoisoned case, the void coefficient remains negative throughout the control region, but it becomes positive enough in the poisoned case to cause the void coefficient for uniform control region void to be a small positive value. It is evident that decreasing the density of poisoned reflector control (uniformly) causes as large an increase in neutron leakage losses as a decrease in thermal neutron absorption.

In assessing the results of Tables 3.6A and 3.6B it should be kept in mind that they apply to a single annulus model. The change of

reactivity for the same change in void in one annulus of ATR is roughly one-third that of the corresponding value in these tables. The delayed group of neutrons from the reflector makes reactivity insertion for prompt criticality considerably greater than the usual value of .72 per cent $\Delta k/k$.

The over-all temperature coefficient of the reference ATR reactor with start of cycle conditions, including 5 g/liter of boric acid in the reflector shim control, has been determined by PDQ calculations and is about $-3 \times 10^{-5} \Delta k/k$ per degree Fahrenheit, Table 3.6C. Calculations on a single lobe model show that the temperature coefficient for the core region is about -8×10^{-5} .

TABLE 3.6C
TEMPERATURE COEFFICIENT ATR REFERENCE REACTOR
5 G/LITER BORIC ACID IN REFLECTOR SHIM

Temperature Degrees F	k_{eff}	Temperature Coefficient $\Delta k/k$ per $^{\circ}\text{F}$	Avg Temperature Coefficient
200	1.13770	-3.6×10^{-5} -2.7×10^{-5}	-3.0×10^{-5}
160	1.13935		
100	1.14117		

3.7 Xenon Instability

The problems of xenon instability can be divided into three types as follows: first, oscillation in a horizontal plane of a single lobe; second, oscillation from lobe to lobe; and third, oscillation in a vertical direction. Calculations using a modification of the method of Wick show¹ that the first type of oscillation, around a single lobe, will not occur.

The reactor is not inherently stable against interlobe oscillations. However, the long period of xenon oscillations and the ability of the control systems to control the interlobe power distribution, Table 3.5A eliminates any problem from the second type of oscillation.

1. R. S. Wick, "Space and Time-Dependent Flux Oscillations (and Instability) in Thermal Reactors Due to Nonuniform Formation and Depletion of Xenon", WAPD-TM-138, August 1958.

Calculations based upon a two-group model with a sinusoidal initial flux distribution indicate that a four foot high core is stable with respect to axial xenon oscillations. However, if the core length is increased significantly, it will definitely be unstable. Further calculations are required to reach a final conclusion on the axial stability. If the core cannot definitely be shown to be stable, provisions should be made for separately measuring the flux levels in the top and bottom halves of the reactor. In addition the control system should have provisions for individual control of a small amount, less than 1 per cent, of reactivity in the two halves. This control is feasible with the chemical reflector shim control if provisions are made for it when the reactor is constructed.

3.8 Neutron Lifetimes

In ATR a significant fraction of the neutrons producing fissions have spent a major portion of their life in the reflector. As a result the reflector contributes what may be considered as an additional group of delayed neutrons. The flux levels required in the tests in ATR can be obtained at reasonable power levels only by designs which have positive void and temperature coefficients in some regions of the reactor. Thus, the delayed neutrons from the reflector are very important to the safe operation of ATR.

These reflector delayed neutrons can be characterized by the fraction of all fissions that they produce and by their average lifetime. In general the type of reflector control does not greatly influence the number of neutrons from the reflector for equal worth of control. However, the average lifetime is dependent not only upon the reflector but also on the type of control used to produce a given change in reactivity.

The importance weighted lifetimes of all neutrons in the reactor have been found for a number of control and reflector conditions by a perturbation method for a one-lobe model. The assumptions are made that all neutrons producing fissions can be considered as having spent almost all of their life in either the reflector (reflector neutrons) or in the core (core neutrons); that all changes in reactivity from reflector control result from reduction in the number of neutrons returned from the reflector; that poisoning the core changes only the thermal utilization and not the relative number of core and reflector neutrons; and that changes in reflector conditions do not influence the average lifetime of core neutrons and vice-versa. These assumptions together with supplementary calculations to determine the fraction of neutrons returned from the reflector when it is unpoisoned permit calculation of the fraction of fissions produced by reflector neutrons and their average lifetime. The results are summarized in Table 3.8A. The number of reflector neutrons is smallest and their lifetime is shortest when the reflector shim control is producing the largest reduction in reactivity, which corresponds to conditions at the start of a cycle. From Table 3.8A it is seen that control by blades or poisoning of a narrow annulus in a reflector results in the longest lifetime for the reflector neutrons under all control conditions. It is also apparent that reactors with most of the reflection from D_2O have an inherently longer lifetime than beryllium reflected reactors.

From the effect on the lifetimes of poisoning the entire reflector it is seen that extraneous absorbers must be kept out of the reflector if the maximum benefit is to be realized from reflector neutrons.

In ATR the fraction of reflector neutrons will be considerably smaller than for the one-lobe model to which the results of Table 3.8A apply. The lifetimes of the reflector neutrons for ATR will be slightly longer because not all of the neutrons return through the control. From comparison of one lobe and ATR calculations of reflector worth it is estimated that at startup, which is the most unfavorable condition for neutron lifetime, at least 10 per cent of the neutrons will be reflector neutrons and that the average lifetime of these reflector neutrons will be approximately 800 μ sec for control by a poisoned H_2O annulus.

TABLE 3.8A

EFFECT OF REFLECTOR SHIM CONTROL ON NEUTRON LIFETIME

Type of Reflector Shim Control	Change in k_{eff} Produced by Control	Avg Lifetime all Neuts μ sec	Avg Lifetime Refl Neuts μ sec	Avg Lifetime Core Neuts μ sec	Percent Core Neuts	Percent Refl Neuts
Clean	0	370	990	35	65	35
Entire Refl	-.03	270	770	35	68	32
Entire Refl	-.13	120	390	35	76	24
Entire Refl	-.24	54	160	35	85	15
Blade	-.03	340	990	35	68	32
Blade	-.13	260	970	35	76	24
Blade	-.24	180	1000	35	85	15
Core Poison	-.03	370	990	32	65	35
Core Poison	-.13	360	960	28	65	35
Core Poison	-.24	350	930	25	65	35
Be + 10% H_2O	-.03	96	230	35	68	32
Be + 10% H_2O	-.13	55	120	35	76	24
Be + 10% H_2O	-.24	34	47	35	85	15
1/2 in. H_2O	-.03	320	930	35	68	32
1/2 in. H_2O	-.13	230	850	35	76	24
1/2 in. H_2O	-.24	130	670	35	85	15

3.9 Gamma Heating

Gamma heating rates have been calculated in various components of ATR using an IBM 704 program developed by Internuclear Company. Briefly, the program performs a point to point numerical integration over the source volume. Differential volume sources are treated as point sources and attenuated as:

$$\Phi(E) = \frac{S_v(E)}{4\pi R^2} \left[A_1(E) e^{-[1 + \alpha_1(E)] b(E)} + A_2(E) e^{-[1 + \alpha_2(E)] b(E)} \right]$$

where

$\Phi(E)$ = radiation flux at the detector point arising from source
 $S_v(E)$, Mev/cm²-sec.

$S_v(E)$ = radiation source strength of energy group E created in
differential volume under consideration, Mev/cc-sec.

R = distance between differential volume source and detector, cm.

$A_1(E)$, $A_2(E)$, $\alpha_1(E)$ and $\alpha_2(E)$ = curve fitting parameters for the NDA
gamma buildup factors for energy.

$$b(E) = \sum_{i=0}^i \mu_i(E) t_i$$

$\mu_i(E)$ = cross sections of the i^{th} shield material for energy group E,
cm⁻¹ (absorption coefficient for gammas and removal cross
sections for neutrons).

t_i = distance traveled through the i^{th} shield material, cm.

The program allows for a non-uniform source distribution and multi-energy groups.

The core gamma energyspectrum and magnitude was calculated considering prompt fission gammas, fission product decay gammas, fission product decay gammas plus capture gammas in U-235, Al and H₂O.

The energy spectrum of the prompt fission and fission product gammas were obtained from numerical integration of recent Oak Ridge data presented by Peelle, et al.¹ Gamma energy spectra for all other system

1. R. W. Peelle, W. Zobel, T. A. Love, "Measurement of the Spectrum of Short-Lived Fission Product Decay Gamma Rays Emitted from a Rotating Belt", ORNL-2081, 1956.

materials were obtained from the work of Deloume¹ or Bartholomew and Higgs².

The average rate of gamma heating in various regions in ATR at a reactor power of 250 Mw are listed in Table 3.9A. The maximum value in any region is about twice the average value.

TABLE 3.9A

GAMMA HEATING IN ATR

<u>Region</u>	<u>Avg Gamma Heating Rate Watts per Gram</u>
Inpile tube	11
Flux trap moderator	14
Core	17
Inner reflector	6
Outer reflector	1
Pressure vessel	0.01

1. F. E. Deloume, "Gamma Ray Energy Spectra From Thermal Neutron Capture", APEX-407, 1958.

2. G. A. Bartholomew and L. A. Higgs, "Compilation of Thermal Neutron Capture Gamma Rays", Chalk River Report No. CRGP-784, 1958.

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FIG. 3.2 A
SCHEMATIC LAYOUT OF ATR

1. OUTER REFLECTOR	10. CENTER EXPERIMENT
2. INNER REFLECTOR
(REFLECTOR SHIM CONTROL)	11. CENTER FUEL REGION
3. OUTER LOOP REFLECTOR	12. CENTER EXPERIMENT
FLUX-TRAP MODERATOR
4. LOBE FUEL REGION	13. NECK FILLER REGION
5. SEPARATOR RING	14. NECK SHIM CONTROL REGION
6. LOBE FLUX-TRAP MODERATOR	15. NECK SHIM CONTROL REGION
7. EXPERIMENT PRESSURE VESSEL	16. SAFETY BLADE
8. LOBE EXPERIMENT	17. EXTERIOR EXPERIMENT
FLUX-TRAP MODERATOR
9. EXTERIOR EXPERIMENT	

FIG. 3.3A

MODEL FOR PDQ REACTOR PHYSICS CALCULATIONS

FIG. 3.4A

MIDPLANE FAST NEUTRON FLUX IN TESTS VERSUS OPERATING TIME

FIG. 3.4B

MIDPLANE THERMAL NEUTRON FLUX IN TESTS VERSUS OPERATING TIME

FIG. 3.4C
BURNUP OF FUEL IN EXPERIMENTS VERSUS OPERATING TIME

FIG. 3.4D
MAXIMUM POWER DENSITY IN CORE VERSUS OPERATING TIME

FIG. 3.5A
CORE POWER DISTRIBUTION VERSUS OPERATING TIME

FIG. 3.5B
PER CENT FUEL BURN-UP IN CORE LOBES VERSUS OPERATING TIME

FIG. 3.5C
 k_{eff} VERSUS OPERATING TIME

FIG. 3.5D
 k_{eff} -MIDCYCLE SHUTDOWN AND RECOVERY

P.P.CO.-C-3017

FIG. 3.5E
CORE POWER DISTRIBUTION—MIDCYCLE SHUTDOWN & RECOVERY

FIG. 3.5F
MIDPLANE FAST NEUTRON FLUX IN TESTS—MIDCYCLE SHUTDOWN & RECOVERY

P.P.CO. - B-3019

FIG. 3.5G

MIDPLANE THERMAL NEUTRON FLUX IN TESTS-MIDCYCLE SHUTDOWN & RECOVERY

4.0 ENGINEERING DESCRIPTION

Section 2.0 presents the core design for ATR, and Section 3.0 discusses its reactor physics. It is the objective of this part of the report to give an engineering description of the entire plant, beginning with a discussion of the reactor itself.

4.1 Reactor

See Figures:

4.1A - ATR Four-Lobe Reactor Structure - Vertical Cross Section

4.1B - ATR Four-Lobe Reactor - Horizontal Cross Section

4.1C - ATR Four-Lobe Reactor - Top Plan

4.1D - ATR Four-Lobe Reactor - Reactor Vessel Top Head -
Top Plan

4.1E - ATR Four-Lobe Reactor - Reactor Vessel Top Head -
Cross Sectional Elevation "A-A"

4.1F - ATR Four-Lobe Reactor - Sub-Pile Room - Horizontal
Cross Section

4.1G - ATR Four-Lobe Reactor - Sub-Pile Room - Arrangement

The four-lobe reactor resembles ETR in the arrangement of reactor, canal, etc., except that special consideration was given to experimental facilities, experiment handling, and meeting much more exacting flux specifications. The reactor core is physically smaller than that of ETR. ATR consists basically of four flux traps in a cloverleaf arrangement of circular segment fuel assemblies. This arrangement provides a reactor with nine excellent experimental facilities.

The following sections give a brief description of the pressure vessel, active lattice and reflector, loop and capsule facilities, experiment handling, fuel handling, reactor structure, and the process piping.

4.11 Pressure Vessel

The multi-diameter reactor vessel, Fig. 4.1A, is vertically mounted and has a spider-shaped top head with an integral shielding sleeve for loop experiments, inlet and outlet cooling water openings, a flow distributor for inlet coolant water, a rotary fuel handling rack built as a part of the flow distributor, a discharge chute, supports for reactor grid plate and experimental in-pile equipment, and a bottom head which incorporates openings for various drive rods and loop piping.

The reactor vessel is to be constructed of stainless steel in accordance with appropriate ASME Codes. Design is to be based on operating pressures of 300 psig and temperatures in the range of 200°F. The upper section of the pressure vessel is about 12 ft in diameter and contains facility nozzles and supports for experiment flow tubes, flanged nozzles for capsule leads, a flow distributor, and the two 24 in. inlet cooling water lines. A rotary fuel rack is also provided for passing fuel elements around the reactor to the discharge chute. The lower section, approximately 6 ft 9 in. in diameter, contains the reactor core, and a grid plate for support of fuel elements, core filler pieces, shim control devices, safety rod guides, and the D₂O reflector tank. Suitable baffling is provided below the grid plate to divide the flow from the fuel elements in order to permit individual flow and temperature-rise measurements for each lobe. Four 16 in. cooling water outlet lines are located below the grid. The over-all length of the pressure vessel is about 32 ft.

The top head is fabricated in the form of a large spider. This spider contains four fueling ports for access into the vessel. The central area of the vessel head contains loop facility penetrations and shielding both above and below the head. The shielding protects personnel working adjacent to the head during the periods that experiment samples are being drawn into the loop cask. Experiment instrumentation leads and tubing run from the top of the in-pile tube penetrations downward to the top of the vessel head and radially outward to the edge of the head via protective conduit runs located over the arms of the spider as shown in Fig. 4.1D. After reaching the edge of the reactor vessel, leads may be routed downward into the nozzle trench. Thus, the leads can run into the experiment cubicles via the same passages that are provided for top-re-entrant or through-loop type experiments.

The vessel bottom head contains drive-rod penetrations for the regulating rods, the safety rods, and the fission and ion chambers. Loop facility penetrations in the top head and in the bottom head should be as large as possible, consistent with good design practices regarding working clearances, seals, in-pile tube removal and insertion, etc. Actual sizing of these penetrations will probably be set by the reactor bottom head, since the safety rod drives, including the holes labeled "possible shim rod drives," also penetrate this head in the vicinity of the in-pile tube penetrations. Top and bottom head penetrations should contain double-gland or equivalent seals that allow pure process water to be introduced between the glands. The nine re-entry loop pipes also enter this head through similar seals (see Fig. 5.1B). Seals should be designed in a manner which will permit re-packing during a normal reactor shutdown, and must be capable of operating reliably in their environment.

4.12 Active Lattice and Reflector

Circular segment fuel elements (see Fig. 4.3A - Circular Segment Fuel Element, Fig. 4.3B - Plan View of Fuel Element for ATR, and Fig. 2.0A - Core Cross Section ATR Four-Lobe Reactor) are used in this reactor. They are supported by a grid plate with sockets for the fuel element side plate extension. Surrounding the core are the borated H₂O shim control system, and the D₂O reflector tank.

For reasons of neutron economy, the area immediately adjacent to and outside of the four lobes must be designed with an absolute minimum of construction materials separating the fuel from the D₂O reflector. This includes the borated H₂O outer control system and the inner wall of the D₂O reflector tank. Of necessity, the vessel design must be based on operating pressures, expected differential pressures, and thermal stresses due to gamma heating. With regard to differential pressures expected between the reactor vessel and the D₂O reflector tank, it appears conceivable to limit this differential pressure to the expected pressure drop across the core by means of a combination pressurizing and pressure-sensing or pressure-balancing system which is tied to both the reactor process water system and the D₂O reflector circulating system.

Referring to Fig. 2.0 - Core Cross Section - ATR Four-Lobe Reactor, a possible arrangement of the borated H₂O outer control and the inner wall of the D₂O reflector tank may be seen. In this instance, the D₂O reflector tank inner wall is reinforced by means of radial ribs extending from the inner wall to the outer wall of the D₂O tank. Also, the outer wall of the D₂O reflector tank is shown with substantial thickness in the above figure. Thermal stress considerations due to gamma heating in the outer wall may dictate the use of circumferential ribs to reduce the wall thickness and hence the thermal stresses in this region.

The core cross section and an alternate core cross section are shown on Figures 2.0A and 2.0B. The core shown on Fig. 2.0A has been discussed above. The core shown by Fig. 2.0B has the same configuration of fuel elements, experimental facilities, and safety rods, but is controlled by means of mechanical control rods and has a beryllium reflector. Combinations of these core arrangements are also possible and may have to be considered. For instance, if the chemical control system appears questionable as a result of the research and development program, it may be necessary to use mechanical control rods as shown on Fig. 2.0B, but it might also be advantageous to retain the D₂O reflector tank shown on Fig. 2.0A. Other considerations might dictate the use of the chemical control system in conjunction with a beryllium reflector.

Four safety rods and two regulating rods are located in the core and D₂O reflector region, respectively, and are driven from

the rod access room below the reactor sub-pile room. Fission and ion chambers are also located in thimbles penetrating the D₂O reflector tank. Drives for these chambers are physically located in the reactor sub-pile room. All drives must be reliable units fully meeting all operating requirements while operating in their environments. Adequate factors of safety are to be incorporated in all designs. Mechanical, electrical-mechanical, or other drives may be considered for this reactor, but the performance of these units must be proved prior to actual use on the reactor.

4.13 Loop and Capsule Facilities

Loop experiment and capsule facilities for the reactor are shown in Figures 2.OA, 4.1A, 4.1B, 4.1C, 4.1D, 4.1E, 4.1F, 4.1G, and 5.1B.

Bottom re-entrant type pressurized water loops (see Fig. 5.1B) extend vertically through the reactor. The top pressure tube closure penetrates the top reactor head and permits removal, insertion, or re-insertion of test assemblies with instrumentation leads intact during one normal reactor shutdown period.

The in-pile tube consists of three concentric pipes or tubes. The inner tube serves as a support for the experiment sample and as a coolant flow channel. The intermediate pipe, of 2 in. inside diameter, contains the loop pressure and also serves as a coolant flow channel. The outer tube contains the insulating gas space surrounding the pressure tube.

In-pile tubes extend vertically through the core and downward through the reactor vessel bottom head. Top and bottom head penetrations are sealed by a double-gland type seal. Pure coolant water is applied between the glands at a pressure slightly higher than that of the reactor vessel. The seals should be designed so that they may be repacked during a normal reactor shutdown.

The in-pile tube bottom head penetrations extend vertically downward (see Fig. 4.1G) and are gently curved outward as they proceed to the catchers located in the sub-pile room floor. These catchers are designed to catch radioactive particulate matter that may be lost from experiment specimens. These extensions might also be fitted with remotely operated vibrators to insure that all solids settle into the catchers. Loop piping is routed from the sub-pile room to the piping corridors via insulated pipes in shielded trenches radiating outward from the center of the sub-pile room. The rod drives extend vertically downward through the sub-pile room floor into a rod access room where they are serviced, removed, etc., in the same manner as in ETR.

Capsule experiment leads, if any, would be brought out through nozzles in the reactor vessel into the capsule trench. The design of the reactor also permits installation of top re-entry loops

and through loops. Piping above the reactor core would be brought out of the reactor vessel via radial nozzles located 10 to 12 ft above the reactor core. Experimental leads would also be brought through the reactor top head penetrations in these cases.

Experiment lead connections should be of a type which will require a minimum of time to make or break connections in the shielded, low pressure terminal boxes located above the reactor vessel top head loop penetrations.

Refueling of the reactor would be accomplished through off-axis ports also located in the reactor top head. This particular arrangement may mean that loop experiment insertion or removal and experiment instrumentation work will be done before or after the refueling operation of the reactor, but this arrangement appears acceptable if all of these operations are optimized in the final reactor and experiment design and are carefully planned and scheduled when these facilities go into operation.

4.14 Experiment Handling

4.141 Loop Facilities

Experiment removal or insertion times will be significantly reduced by utilizing an experiment handling cask and cart which operate on straight rails extending over the reactor and reactor working canal. The cask will be supported on a bridge and trolley arrangement which will permit positioning over any of the nine in-pile pressure tubes and also over the canal discharge opening. The cask is designed to handle two test assemblies simultaneously in order to permit interchange of irradiated test assemblies in one operation. The trolley is to have an elevating mechanism which will permit the cask to be lowered onto the removal sleeve and to be raised to clear the reactor structure. Cask and trolley are to be mechanically driven and to have scales, indexing pins, etc. for determining position over the reactor and canal. Provision for a water circulation system must be included in the cask design, since it may be necessary to keep the test assembly under water during removal and discharge operations. The cask, trolley, and bridge girders are to be removable with the 50 ton bridge crane operating over the reactor area.

In order to more fully establish design criteria for the experiment handling cask, handling tools, the in-pile tube, and other items, a stepwise procedure for removing a sample from an in-pile tube is included in the following material:

1. The loop will be depressurized, in-pile isolation valves closed, and the in-pile section vent opened. The in-pile tube will be full of water.

2. Disconnect all instrument leads from both inside and outside of the low pressure terminal box. Move extension leads out of the way.

3. Remove the low pressure terminal box. Tape all sample instrument leads together in a vertical position and/or install a protective cover over leads in place of the terminal box.

Note: The outside diameter of this cover must be smaller than the inside diameter of the cask grapple.

4. Remove the top closure locking ring which holds the top closure plug in place.

5. Install the removal sleeve over the outside of the pressure tube top head penetration.

6. Position the cask over the in-pile tube with approximately 2 ft of space between the cask and the top of the pressure tube. Open the bottom door of the cask.

7. Lower the grapple through the cask and attach it to the end of the top closure plug.

8. Pull the test assembly up a few inches to be sure it is not bound in the flow tube, then carefully lower it.

9. Lower the cask to contact the removal sleeve. A water seal should now be effected.

10. Fill the cask and removal sleeve with water and check for leakage between removal sleeve and cask.

11. Raise the test assembly by means of a lifting mechanism mounted integrally with the cask which allows the operator positive control and indicates the amount of tension on the cable. This allows the operator to immediately detect binding or catching of the test assembly.

12. The sample will be raised until the attached grapple reaches a stop which indicates that the bottom of the sample is safely above the cask door.

13. Close the cask door. The cask may need a circulating water system to remove sample afterheat. If so, put it in operation at this time.

14. Open the vent valve on the removal sleeve, drain the water from the sleeve and check for leakage through the cask door.

15. Raise the cask enough to be sure the cask doors are not leaking water.

16. Move the cask to the discharge position over the test assembly transfer tube using the cask positioning bridge and trolley. The test assembly transfer tube must have provisions for maintaining the water head in the cask until the test assembly has been lowered into the canal and must support the test assembly at the top closure plug after removal from the cask.

If the test assembly is scheduled for shipment to some other facility, the cask only would be handled with the 50-ton overhead crane which operates over the reactor area.

17. Discharge test assembly to tube.

18. The in-pile tube assembly is now ready for insertion of another test assembly.

Note: The removal cask has provisions for simultaneously handling two irradiated test assemblies, i.e., two test assemblies could be removed from the reactor at one time. The dual removal feature would also permit the interchange of an irradiated test assembly in the reactor with another scheduled for reinsertion. The test assembly freshly removed from the loop could then be shipped or moved to the experimenters' canal for inspection.

19. Insertion of an irradiated test assembly into the loop would be essentially the reverse of the above procedure.

20. Unirradiated test assemblies could be installed using the 2-ton handling crane which operates over the reactor and working canal.

4.142 Capsule Facilities

A capsule handling cask is supported on circular rails on the top of the reactor structure and will also operate over and discharge into the reactor working canal. This cask may also be handled with the 50-ton bridge crane.

4.15 Fuel Handling

Fuel handling (Figures 4.1A, 4.1G, and 4.1C) is accomplished through fueling ports in the top head. A rotary fuel storage rack is built inside the flow distributor so that fuel may be passed around the reactor, under the loop flow and lead tubes, and under the top head spider arms to the discharge chute.

One important feature of the fuel storage rack is its ability to hold an entire fuel element charge, thereby making it possible to refuel for an emergency shutdown without having to discharge each element into the working canal until the next regular shutdown.

4.16 Reactor Structure

Figures 4.1A, 4.1B, and 4.2D show the structural features around the reactor.

In addition to the rails which support the experiment handling casks, shielded tunnels, cubicles, etc., are located in the reactor structure and around the periphery of the reactor vessel as necessary to service capsule nozzles, borated H₂O piping, D₂O piping, and top re-entrant or through loop flow tubes. Piping corridors and suitable reactor vessel support structures are also provided. The outer thermal shield is shown outside the reactor pressure vessel and inside the thermal shield vessel where it is cooled by process water. This arrangement offers greater flexibility of choice in reflector materials, and it will handle any of the design alternates being considered. Joining the reactor on the north side is a working canal with transfer tubes for fuel and experiment handling and a viewing window for underwater inspection of fuel assemblies, experimental equipment, etc. The long experiment transfer tube is slotted its full length so that discharge into the canal is possible.

4.17 Process Piping

A main header from the process water area (Fig. 4.2D, Reactor Building Section A-A) brings cooling water to the reactor vessel by way of a piping tunnel beneath the canal. The two 24 in. inlet lines then run to the flow distributor in the vessel where the water is dispersed through the upper section of the vessel. The water flows downward through the core and leaves the vessel through four 16 in. exit water lines. The process water then returns to the process water area by way of the piping tunnel.

FIG. 4.1 A
ATR FOUR-LOBE
REACTOR STRUCTURE
VERTICAL CROSS SECTION

FIG. 4.1B
 ATR FOUR-LOBE REACTOR
 HORIZONTAL CROSS SECTION
 SECTION A-A FIG. 4.1A

FIG 4.1C
 ATR FOUR-LOBE REACTOR
 TOP PLAN
 SEE FIG 4.1A

FIG 4.1D
ATR FOUR-LOBE REACTOR
REACTOR VESSEL TOP HEAD - TOP PLAN

FIG. 4.IE
 ATR FOUR-LOBE REACTOR
 REACTOR VESSEL TOP HEAD
 CROSS-SECTIONAL ELEVATION "A-A"

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FIG. 4.1F
A T R F O U R - L O B E R E A C T O R
R O D A C C E S S R O O M
H O R I Z O N T A L C R O S S S E C T I O N
S E C T I O N B - B F I G . 4 . 1 A

PLAN VIEW SUB-PILE ROOM

SECTION C-C

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FIG. 4.1G
ATR FOUR-LOBE REACTOR
SUB - PILE ROOM
ARRANGEMENT

4.2 Reactor Building Complex

Design philosophy leading to the location and relationship of the reactor, critical facility, storage canal, and the experimenters' section of the canal is discussed in the material following:

1. Common canal access is provided for all of these facilities since it appears likely that there might be considerable interchange of loop experiments, capsules, fuel elements, etc., between any or all of the above facilities. Under such conditions, canal transfer appears more attractive than cask transfer when considering such things as time consumed in effecting a given transfer, likelihood of physical damage during transfer, and other considerations.

2. Each facility should nominally be capable of operating regardless of conditions existing in adjoining facilities. Toward that end, each facility and its access routes would be isolated from every other facility.

3. Actual floor space provided for each facility should take into consideration the experimental programs and the operational and maintenance requirements for that facility.

The above design considerations led to the layouts shown on Figures:

4.2A - Reactor Building - First Floor Plan

4.2B - Reactor Building - First Basement Plan

4.2C - Reactor Building - Second Basement Plan

4.2D - Reactor Building - Sectional Elevation A-A

4.2E - Reactor Building - Building Elevations

As indicated on the above layouts, the reactor, critical facility, and experimenters' section of canal are all located on, and connected to, a common storage canal.

The north side of the reactor building contains the storage canal area which would be used to store hot experiments, fuel elements, and experimental equipment; to load cut fuel and experimental equipment into shielded carriers; to transfer experimental equipment into shielded carriers; and to transfer experimental equipment, fuel elements, etc., between facilities located in the reactor building. Gamma irradiations might also be conducted in this storage canal. The entire storage canal area would be serviced by a 50-ton bridge crane.

The extreme west end of the canal area contains the experimenters' section of canal. This canal is separated from the main storage canal

by means of a bulkhead. Hot experiments and experimental equipment could be visually examined and remotely worked on from the top of the canal parapet or from behind shielding windows located below water level on the south side of the experimenters' canal.

In the center section of the reactor building is the reactor area. A reactor working canal extends from the storage canal through the north wall of the reactor area to the reactor structure. Transfer from the reactor working canal to the storage canal would be accomplished by means of a mechanical cart or other device located in the bottom of the connecting canal and operated from the canal parapet on both sides of the north isolation wall. This isolation wall will extend a nominal distance below water level to form an air seal between the reactor area and the storage canal area. The reactor area is serviced by a 50-ton bridge crane running in a north-south direction and by a 2-ton handling crane located above the 50-ton crane and operating over the reactor and working canal. A freight and passenger elevator, shift supervisor's office, reactor control room, and HP issue room are located on the south wall of this area. Truck access to the reactor area is provided through a truck door located on the south wall of the reactor area. The reactor area, the critical facility area, and the storage canal area are to be designed so that air or surface contamination in one area can be contained in that area and will not affect operations in any of the other reactor building areas.

The area east of the reactor area is the critical facility area and contains a 24 ft long canal in which the four-lobe critical facility would be located. The critical facility canal is connected to the storage canal by a short canal extension. Transfer of heavy materials through this connecting section would be accomplished by means of a mechanical cart or other device located on the bottom of the connecting canal and operated from the canal parapet on both sides of the north sliding isolation curtain. The critical facility is served by a 15-ton bridge crane running in a north-south direction.

A decontamination room, rest room, and an HP office are located just south of the critical facility offices. The heating and ventilating equipment area is located on the second floor above these offices.

The office area west of the reactor area provides additional office and service space for functions essential to the operation of the reactor and experimental program.

The two basement levels in the reactor area of the reactor building are utilized for experimental cubicles, consoles, sample areas, etc.; for the process equipment directly associated with the borated H₂O shim control system; and for reactor instrument and instrument repair rooms. Other basement areas are utilized for a counting room, a fuel storage vault, cubicle exhaust equipment, and data reduction equipment.

Location of the reactor process water and other process equipment is also indicated on the above figures, but will be more fully discussed in Section 4.7, "Site, Buildings, and Facilities".

Building floor loads and canal floor loads should be specified at time of final design. In many instances, these loadings have tended to be inadequate in facilities built to date. Representative loadings for various areas might approach the following:

Reactor Area - All Floors	5000 lb/sq ft
Storage Canal Area	5000 lb/sq ft
Critical Facility Area	3000 lb/sq ft
Experimenters' Canal Area	3000 lb/sq ft
All Canals	3000 lb/ sq ft + water load

FIG. 4.2 A
REACTOR BUILDING — FIRST FLOOR PLAN

FIG. 4.2 C
REACTOR BUILDING—SECOND BASEMENT PLAN

FIG. 4.2 D
REACTOR BUILDING—SECTIONAL ELEVATION "A-A"

REACTOR BUILDING SOUTH ELEVATION

REACTOR BUILDING EAST ELEVATION

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FIG. 4.2 E
REACTOR BUILDING — BUILDING ELEVATIONS

4.3 Fuel Element

4.31 General

The fuel element design for the proposed reactor has been based on criteria dictated by the core physics, cooling requirements, and materials technology. The high neutron flux and associated power density in the reactor specifications indicated advanced fuel technology, yet it has been desired to achieve the immediate objectives with a minimum of development effort. An attempt has also been made to select a design offering adequate flexibility for future increases in reactor power. Computations of core physics have produced such parameters for the fuel element design as maximum and average power density, metal-to-water ratio, and critical mass. Heat transfer considerations have provided channel size, coolant velocity, core pressure drop, pressure differential across the fuel plates, and maximum wall temperature. With the maximum wall temperature in the 400 - 500°F range, aluminum technology is applicable. Fuel loading has necessarily been high to accommodate the high power density. Uranium-aluminum alloys of 35 weight per cent uranium have been considered as the present fabrication limit but with 45 weight per cent possible with some development. The rolled "picture frame" plate has been retained as an adequate and well developed fabrication technique. To retain the strength of the hardened aluminum alloy, assembly by brazing has been eliminated in favor of roll swaging or pinning. This technique has been verified by the ETR operating experience.

In general, the fuel element features desirable for optimum heat removal are contradictory to those features required for the best structural characteristics. Heat removal considerations indicate that both the side plates and fuel plates should be thin and that the span of the fuel plates should be large. However, strength requirements dictate that the side plates and fuel plates should be relatively thick and that the unsupported span of the fuel plates should be minimized. Fluid velocities and wall temperatures have a direct effect upon both the heat removal and the structural properties of the element. For example, increasing the nominal fluid velocity from 30 to 40 ft/sec increases the heat removal capabilities by a factor of approximately 1.2, while the lateral core pressure differential is increased by a factor of 1.8. Similarly, increasing the wall temperature by 50°F will increase the heat removal ability of the element by about 20 per cent while the structural strength of the fuel plate is reduced by about 40 per cent. High velocities and high temperatures increase the rate of corrosion of the cladding and therefore necessitate the use of thicker cladding.

From a heat removal standpoint, there are obvious potential methods of improving the element such as decreasing the number of elements per lobe, decreasing the plate thickness and increasing the number of plates per element. However, each of these changes decreases the structural quality of the element and therefore cannot be recommended until sufficient development work is done to establish the complete reliability of the modified element under the specified reactor conditions.

4.32 The Circular Segment Fuel Element

The geometry most nearly meeting the specifications outlined in the previous section appears to be an element composed of circular segment fuel plates (Figures 4.3A and 4.3B). Basically the element consists of curved plates mounted between two side plates to form a 45° segment of a right circular cylinder. The inside radius is 3 in. and the outside radius is 5.5 in. The side plates are extended to form the bottom end box. Each element has nineteen curved plates 0.050 in. thick, mechanically joined to the side plates. Each fuel plate consists of a homogeneous, highly enriched uranium-aluminum alloy core, 0.020 in. thick, completely clad with a 0.015 in. layer of X8001 aluminum alloy which should be sufficient for the corrosion and stress conditions encountered. The overall length of the plates is 49.5 in. and the fuel loading required is 33 weight per cent. Boron is incorporated in the core alloy to compensate for the loss in reactivity due to fuel burnup.

The 20 per cent enriched core loading experience in the MTR indicates that the proposed alloy should be stable under low temperature irradiation. Test samples similar to the proposed fuel plates are presently being tested under typical MTR operating conditions. However the corrosion resistance and irradiation stability of the proposed composition should be verified in in-pile tests and under the temperature and flow conditions specified for the ATR.

Some increase in cost of the circular segment fuel element over the present MTR-ETR type elements can be expected due to the number of dies needed to fabricate the individually sized plates, the incorporation of boron, the special alloy cladding, and the reject rate of the high weight per cent core. Recent ETR experience has shown that the boron distribution can be satisfactorily controlled in production. No increase in price is anticipated if it becomes desirable to grade the fuel radially to flatten the power. A slight increase in cost (i.e., 10 per cent) might be expected for axial grading of fuel.

Other fuel element designs and materials which were considered are discussed in Section 8.6.

4.33 Fuel Element Flow Containment

Because the coolant velocity through the fuel (44 ft/sec) is much greater than the velocity through the adjacent core regions, some provision is needed either in the design of the fuel element or in the reactor structure to effect adequate cooling of the external plates and to establish flow conditions such that no unacceptable lateral pressure differentials exist across these outside plates. If the flow immediately adjacent to the outside fuel plates were allowed to have the same velocity as the flow through the adjacent region, then a lateral pressure differential of approximately 86 psi would exist across the outside plates.

Three methods are suggested for providing a 0.078 in. thick flow channel adjacent to the exterior fuel plates so that the velocity on both sides of these fuel plates is the same and no excess lateral pressure will exist.

1. Where aluminum filler pieces are adjacent to the fuel, as in the center high flux region, these filler pieces will be designed to provide the required flow channel.

2. Where structural members are adjacent to the fuel, as on the outside of the four lobes, these members will be designed to provide the required flow channel for the outer fuel plates. An example of this is the borated outer control tanks.

3. Where a water region is adjacent to the fuel, as in the flux-trap regions in two of the lobes, complete right circular cylindrical shells will be used to separate the fuel region from the adjacent water region and to provide the necessary flow channels adjacent to the external fuel plates.

If provisions, such as those outlined above, are not made to isolate the low coolant velocity regions from the high coolant velocity fuel region and to provide channels which permit high flow rates along the two outside fuel plates, then relatively thick (0.180 in.) non-fueled outer plates will be required on the fuel elements.

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FIG. 4.3A
CIRCULAR SEGMENT FUEL ELEMENT

FIG. 4.3 B

PLAN VIEW OF FUEL ELEMENT FOR ATR

4.4 Reactor Controls and Instrumentation

The instrumentation system may be divided into three categories for purposes of discussion: the control system which utilizes regulating rods and shim control to allow automatic operation of the reactor under preset conditions; the safety system which functions to protect the reactor throughout all conceivable operating situations, intended or accidental; and the process instrumentation to allow the proper processing of the reactor coolants according to the reactor requirements.

4.41 Control System

The control system provides a continuous means of measuring the power level and the neutron flux level in the reactor and a means of controlling the long time variations (shim control) and the short time variations (regulating rod control).

4.411 Power Level Instrumentation

There are two systems for determining the integrated power level of the reactor as shown in Fig. 4.4B. The first is a system of water sample tubes spaced around the reactor to bring a sample of the coolant water from five regions out to a gamma ray detector arrangement. This sample is at the detector within approximately 10 sec after passing through the reactor core so that the activity caused by the neutron reaction $^{16}\text{O}(n,p)^{16}\text{N}$ which decays with a 7.4 sec half life can be measured. By sampling at five symmetrical regions in the reactor an integration of the reactor neutron flux is obtained. In addition, this measurement allows balancing of the neutron flux in the reactor by shim control when experimental loading causes unbalance.

The second system consists of a conventional computation of the reactor power by measurement of the difference in temperature of the inlet and outlet cooling water and multiplying by the coolant flow rate. Several resistance thermometers are used to allow better integration of the inlet and outlet temperatures reducing the effect of any stratification in the coolant water piping.

A differential galvanometer allows a more sensitive indication of the drift in power level than on a normal full scale indication. The system shown in Fig. 4.4A with a carefully controlled zero offset is quite sensitive in this measurement. With batteries as the high voltage supply, the system is also used as a power level indication independent of ac power generation. The chamber for this instrument may be located outside of the thermal shield since it does not require fast response and can thus operate at lower neutron sensitivity.

4.412 Shim Control

A system of shim control is proposed which uses a chemical solution of boron to be added to or taken away from two control regions in each of the four lobes otherwise containing H_2O as shown in

Fig. 2.0A. This type control has an advantage over the fuel follower type shim rod control in that it perturbs the neutron flux symmetrically in the vertical direction. This in turn relieves an otherwise more critical heat generation problem and allows a more uniform neutron flux throughout the length of the experimental tubes.

A chemical shim control system design for the reflector region only demonstrates the problems and possible solutions encountered. Directly outside the core is a one-half inch thick H_2O reflector annulus to which H_3BO_3 can be added or removed to provide a chemical shim control system. A schematic diagram of such a system is shown in Fig. 4.4D. The shim control annulus is divided into four sections to allow differential control of reactivity imbalances from the experiments, to effect power balancing during reactor operation and to allow special distribution of power among the four lobes to achieve a special combination of flux levels. In addition, a similar system of chemical shim control is provided in each of the four neck regions for the purpose of aiding in power balancing and distribution. A double one-half inch I.D. tube system straddling each mechanical safety rod is used with each side a re-entrant tube. This design allows easy replacement of a defective unit from the top.

Due to the high gamma and neutron heating in the borated H_2O tank and neck regions, the H_2O is cooled in an external heat exchanger. A baffle and orifice is provided to force part of the H_2O at high velocity past the inner walls of the borated H_2O tanks to increase the heat transfer coefficient and reduce the tank operating temperature.

For the removal of boron from the reflector stream, a boiler- H_3BO_3 recovery system is used. The recovery system stores the portion of the borated H_2O withdrawn from the reflector stream and distills off the H_2O as a batch process. The high B^{10} content boric acid is then recovered and stored in the injector makeup tanks. If standard boric acid is used, an ion-exchange system would be feasible for removal of boron and radioactive products. In this event boric acid recovery would be needed as the amounts used during each cycle would be quite small.

For addition of the boron to the system, an injector places a metered amount of $130^\circ F$, 100 g H_3BO_3 /liter solution into the main H_2O stream just prior to entrance into the reflector tank region.

The equations describing both the addition and removal of boron are shown below. For removal of boron from the system,

$$\frac{C}{C_0} = e^{-\frac{\beta v_s}{V_R} t},$$

where

- C = concentration of H_3BO_3 at time, t , in g/liter,
 C_0 = initial concentration of H_3BO_3 in g/liter,
 v_s = flow rate of the coolant in ft^3/sec ,
 β = fraction of flow being drawn off,
 V_R = total reflector volume in ft^3 , and
 t = time in sec.

Fig. 4.4E shows the variation in the effective multiplication factor, k_{eff} , for a continuous cleanup rate as criticality is approached from $k_{eff}=0.98$. For a shim solution initially containing 50 g/liter of H_3BO_3 it is required that $\beta v_s/V_R$ equal 0.00335, which is an injection rate of 1.0 gpm, in order to achieve criticality in four minutes after shim solution cleanup begins. Even faster reactor startup times are possible, however, as this rate of change of reactivity does not approach the maximum limit of 10^{-3} in $\Delta k/k$ per sec. It is also noted that this can be an intermittent process instead of a continuous one.

The boron injection system is described by the equation,

$$C = C_0 + \beta(C_a - C_0) \left(1 - e^{-\frac{v_s}{V_R} t}\right)$$

- C_a = H_3BO_3 concentration being injected in g/liter,
 C_0 = initial concentration of the system in g/liter,
 v_s = system flow rate in ft^3/sec ,
 v_a = injection flow rate in ft^3/sec ,
 t = injection time in sec,
 V_R = total reflector volume in ft^3 , and
 $\beta = \frac{v_a}{v_s}$.

This equation is valid only for a single cycle of the reflector system because of the delay time from starting injection until the concentration increase is felt throughout the system. For times longer than the system cycle time a series of successive approximations must be used to determine the concentration of the fluid entering the reflector. The cycle time

effect does not allow injection at a constant rate to maintain a uniform increase in boron concentration. Refinements in injection techniques and mixing in the external system should approximate a uniform rise, however.

It is desirable to have similar sensitivity of control for the addition of boron as occurs for the removal of boron in the reflector. In Fig. 4.4F the value of v_a/v_s required to initiate a 10^{-3} /sec rate of change of reactivity for the operating reflector boron concentrations is obtained. In order to maintain this rate v_a/v_s will have to increase exponentially at approximately the inverse rate of change of k_{eff} with boric acid concentration. A typical curve of k_{eff} vs. boric acid concentration is shown in Fig. 4.4G.

The preliminary design calculations show that this type of reflector shim system does not perturb the test assembly flux unduly and gives a gradual and stable reactor control. The engineering development work should be directed to provide a smooth, reliable, and proper rate of addition of pure H_2O and concentrated H_3BO_3 solution. The feasibility of this system has been demonstrated in the BORAX Facility located at NRTS.

The boron chemical shim control system is sufficiently flexible to allow adequate compensation for changing reactor conditions. The rate of change of reactivity available in the system is greater than that produced by the maximum rate of change of xenon or temperature under the worst operating conditions. Controlling of each quadrant individually allows suppression of any oscillation in the flux that might be caused by the degree of coupling between the four lobes. It also permits adjusting the five major loops individually to different operating flux conditions.

4.413 Regulating Rod Control

The regulating rod is a vertically operating rod located in the D_2O reflector as shown in Fig. 2.0A. The rod inserts and withdraws neutron absorbing material to damp small reactor power fluctuations as dictated by the servo control system. The servo system which allows automatic control of the reactor power is shown in Fig. 4.4H. The ionization caused by neutrons in the compensated ion chamber allows a current proportional to the neutron flux to flow through the motor operated rheostat. The rheostat acts as a set point to produce a voltage from the ion current that is balanced against a fixed voltage in the servo amplifier. As the rheostat is varied a different power level is determined and the hydraulically driven rod allows a correction to be made to bring the voltage into balance. Two decades of power level are controlled by the servo system.

It is possible that with a chemical shim control system as described above a mechanical regulating rod may be unnecessary. To determine this an investigation into the mixing problems, the time response and the practical processing of the chemical solution is necessary. This proposal is made including a mechanical regulating rod.

As discussed in the kinetic calculations in the hazards section of this report, there is an added measure of safety in the control of the water reflected reactor. Those neutrons entering the reflector return to the active core with some finite delay in the order of a milli-second. The effect is similar to that of the delayed neutrons which is the mechanism allowing control in a thermal reactor. Only in this case the percentage of neutrons delayed is some 30 per cent instead of the 0.7 per cent normally available. The result is that the performance of the servo system can be relaxed over a thermal reactor that does not have heavy water reflection.

It is noted that the boron in the shim reflector can react on the worth of the regulating rod due to the close proximity of the two. However, at criticality and power where the regulating rod is operating, the amount of boron in the shim reflector will be reduced and the effect should be minimized.

4.42 Safety System

The safety system receives its activation from fission chambers and gamma-compensated ionization chambers, which, either separately or in combination, provide a continuous indication of the neutron level in the reactor from the cold clean reactor (with neutron source) to greater than 150 per cent of full power rating. These channels are adjusted appropriately to give the proper shutdown of the reactor for unsafe operating conditions in whatever region of neutron level the reactor is operating.

4.421 Startup Instrumentation

The startup instrumentation consists of two channels of counting rate instruments to count the fissions in two fission chambers. The output from the log count rate meters as shown in the block diagram in Fig. 4.4C is recorded. On the recorder there are microswitches set so that the control system will not allow the reactor to start up without having at least one of the count rate channels reading on-scale. These channels allow observation of the neutron flux in the cold clean reactor region with a reasonable size neutron source from which to multiply.

4.422 Mechanical Safety Rods

In ATR the safety rods are blade type curtains which move between adjacent fuel sections in the stem of each of the four lobes. The rods are driven from below the core by a system similar to that in use in ETR. During reactor operation the safety rods are withdrawn completely to a position one foot above the core. In the event of a scram signal, the holding magnets are de-energized, releasing the rods which are accelerated by gravity and core pressure into the reactor.

The safety rod drop time is found from the solution of the acceleration equation to be

$$t = \frac{w}{KA \sqrt{\frac{2g}{K} \left(\Delta p + \frac{w}{A} \right)}} \cosh^{-1} \left(2 e^{\frac{KAL}{w}} - 1 \right) ,$$

where

w = the rod weight in lb,

A = the rod cross sectional area in ft²,

K = a conversion factor,

Δp = the pressure drop across the core in psf, and

L = the length of the core over which pressure drop occurs in ft.

Neglecting dashpot effects at the end of the rod travel, the safety rod drop time is approximately 0.47 sec with a Δp of 75 psi. Assuming a negligible delay time in the instruments and a release time for the magnets of 0.05 sec, the total time required to introduce the complete shutdown reactivity is 0.52 sec.

The shim control system is adjusted to have enough negative reactivity effect that at no time will the reactors go critical on withdrawal of the safety rods during startup. To account for the excess reactivity needed for burnup and the possible additional positive reactivity that might occur from fuel experiments, a value of effective multiplication of 0.98 has been chosen for the cold clean reactor when the safety rods are completely withdrawn.

4.423 Safety Instrumentation

The safety instrumentation for the reactor makes use of many of the desirable features in the MIR. The block diagram of the safety instrumentation is shown in Fig. 4.4I. The signals from three level channels and two period channels originating from five separate chambers are tied to a common sigma bus. This sigma bus acts as an input signal to the nonlinear magnet amplifiers. The magnet amplifiers have the characteristic that as the input from the sigma bus goes either up above a certain voltage or down below a certain voltage, the output current to the magnets goes down. Thus, for a reactor power increase and the resulting sigma bus voltage increase or for shorting of the sigma bus resulting in a voltage decrease, the magnet current decreases sufficiently to release the safety rods into the reactor. This system is commonly referred to as the "Fast Scram" or "Electronic Scram"

instrumentation since it can occur in the millisecond range and does not require any mechanical relay operations. The "Slow Scram", "Setback", and "Reverse" are usually initiated by recorder level switches using relays. A list of typical "Corrective Signals" to be used are given in Table 4.4A.

TABLE 4.4A
CORRECTIVE SIGNALS

Parameter	Setback Signals	Reverse Signals	Scram Signals
Neutron level	$>1.1 N_F^*$	$>1.2 N_F$	$>1.3 N_F$
N-16 power level	$>1.1 N_F$		$>1.3 N_F$
Water power	$>1.1 N_F$		$>1.3 N_F$
Water temperature rise	$> 105\% \text{ Normal}$		$> 150\% \text{ Normal}$
Water flow	$>_{\pm} 10\% \text{ Normal}$		$>_{\pm} 20\% \text{ Normal}$
Water pressure	$>_{\pm} 10\% \text{ Normal}$		$>_{\pm} 20\% \text{ Normal}$
Manual operation	Yes	Yes	
Neutron level (manual)		$>3 N_L^{**}$	
Reactor period		$<5 \text{ sec}$	$<1 \text{ sec}$
Neutron level (Reactor period $<1 \text{ sec}$)			$> N_L$
Loss of instrument voltage			Yes
Purchased power frequency out of limits			$>3 \text{ cps}$
Loss of purchased power			Yes
Loss of relay bus voltage			Yes

* N_F = Full Power

** N_L = 1% Full Power

The Setback is used to bring about a gradual reduction of operating power with the occurrence of an abnormal condition, not necessarily unsafe, but which is either a precursor of an unsafe condition or that which would invalidate some standby protection in the event of a real emergency. The setback is accomplished by the automatic lowering of the control point of the servo system and persists only so long as the condition which provoked action remains. The reactor is always essentially critical. When the trouble has cleared, the servo will respond to the operator's demand for return to the original or desired servo setpoint.

The Reverse is more drastic in that all control elements are driven in the direction of decreasing reactivity and continue this action until the condition has cleared and the operator has removed the reverse command. Depending upon how long the reverse lasts, the reactor may be subcritical by several per cent; the withdrawal of shim rods must be initiated by the operator to restore the reactor to normal operation. Whether this can be done in any particular case will depend upon the neutron level at the time increasing the reactivity is attempted by the operator. If the flux has fallen below some safe, arbitrary value, the system will demand the operator go through the entire startup procedure. Obviously the reverse is used for conditions which require drastic action, such as a 5 second period.

The Scram is the final and most dramatic corrective action which may be taken by the control system and results in complete shutdown. The ultimate safety of the reactor depends on the existence of numerous independent means for achieving the scram, so that the probability of simultaneous failure of all is made vanishingly small.

4.424 Auxiliary Instrumentation

Fission Product Monitoring System¹ - In a high-flux water-cooled reactor it is an important consideration to detect contamination of the system from fresh fission products. The detection and localization of such contamination is necessary from an operational viewpoint in order to make decisions regarding reactor operating level, disposition of effluent water, and the operation of experimental facilities.

The instrumentation must discriminate against the high "noise level" caused by radiation from activated impurities in the water, by entrance of radioactive nuclei from structural materials and deposition of radioactive material on the surface of the water system. This problem is alleviated by using fission product iodine isolation by means of ion exchange resins and gamma energy discrimination as indicated in Fig. 4.4J. By looking at the short-lived iodine products, past history is not a major factor in the measurement.

1. R. L. Heath, "Fission Product Monitoring in Reactor Coolant Streams", IDO-16213, January 1, 1956.

Boron Concentration Instrumentation - The startup of the reactors requires that a 70 per cent saturated solution of H_3BO_3 be present in the H_2O shim control reflector so that the reactor does not go critical on withdrawal of the safety rods. A test should be available to assure the operator that this condition does exist. A primary measurement of the neutron absorption can be made as shown in the block diagram in Fig. 4.4K. A polonium-beryllium neutron source located adjacent to a thermal neutron detector will emit neutrons into the shim reflector region around it. The number of thermal neutrons reflected back to the detector is a function of the concentration of boron in the heavy water. The system must be calibrated at several known points as the measurement is a non-linear relationship with concentration.

There is a boron instrumentation system on each of the separate segments in the reactor that have variable boron concentration for control purposes. This information is displayed on a recorder calibrated in concentration and coupled into the control system for startup control.

Health Physics Instrumentation - The health physics instrumentation included here is a remote monitoring constant air monitor which has sampling stations in critical regions throughout the reactor buildings. These stations indicate and alarm at a central location in each reactor building. Other health physics instrumentation, such as the portable instruments to measure normal radiation types, will also be available.

4.43 Process Instrumentation

All important process variables are recorded and displayed in the appropriate locations. Since this instrumentation is fairly standard and straightforward as related to reactor operation and control, it is not detailed here. The cost of this instrumentation is, however, included in the final cost estimate.

FIG. 4.4A
DIFFERENTIAL GALVANOMETER

FIG. 4.4B
POWER LEVEL INSTRUMENTATION

FIG. 4.4C
STARTUP INSTRUMENTATION

FIG. 4.4D
TYPICAL BORATED H₂O REFLECTOR SYSTEM
FLOW DIAGRAM

FIG. 4.4E
TYPICAL CURVE OF k_{eff} vs. TIME FOR H_2O DILUTION OF BORATED REFLECTOR

FIG. 4.4F
TYPICAL BORON INJECTION RATE YIELDING
 $dk/dt = 10^{-3} \Delta k/k\text{-sec}$ AT A GIVEN SYSTEM CONCENTRATION

FIG. 4.4G
TYPICAL k_{eff} vs. CONCENTRATION FOR BORATED REFLECTOR SYSTEM

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FIG. 4.4H
POWER REGULATING SYSTEM

FIG. 4.4I
REACTOR SAFETY INSTRUMENTATION

FIG. 4.4J
FISSION BREAK INSTRUMENTATION

FIG. 4.4K
BORON CONCENTRATION INSTRUMENTATION

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4.5 Reactor Cooling and Heat Dissipation

Moderately pressurized light water flowing at relatively high rates is used to remove the heat generated in the fuel elements and associated hardware in the reactor. Heat is eventually rejected to the atmosphere through forced-draft cooling towers. Shell-and-tube heat exchangers are used to transfer heat from primary to secondary cooling water.

4.51 Reactor Cooling

The operating conditions of the reactor primary coolant system were established by means of a modified hot spot - hot channel analysis together with three other criteria:

1. The maximum pressure differential across any fuel plate at operating conditions is equal to or less than 5 psi.
2. The maximum allowable wall temperature, using the hot spot - hot channel analysis, is the saturation temperature at the hot-spot pressure plus 20°F.
3. The maximum steady state wall temperature with Type X8001 aluminum-alloy cladding is not to exceed 500°F.

The lateral pressure differential across any fuel plate must be limited to insure operability of the fuel element under dynamic conditions. Static pressure testing of several MTR and ETR fuel elements which had fuel plates attached to the side plates by different methods has shown, in general, that about 10 psi is required to obtain a 0.010 in. deformation in the fuel plates. It has also been shown that developing lateral differential pressures of this order under dynamic conditions causes closure of some of the coolant channels between fuel plates. A lateral pressure differential of 5 psi was selected as the maximum permissible since the high fuel plate temperature substantially decreases the yield strength of the aluminum alloy, and operation of the ETR with pressure of about this magnitude has not resulted in failure of the fuel elements.

In both the MTR and ETR the maximum calculated wall temperature is allowed to exceed the saturation temperature at the hot spot by about 20°F. This is obtained through essentially the same modified hot spot - hot channel analysis. Section 8.7 of this report explains this analysis in detail. The hot spot - hot channel and power distribution factors that were used for the reactor are summarized in Table 4.5A.

The factors cited are based on operating data, fuel element inspections, and chemical analysis of the MTR and ETR fuel elements. Therefore, they are considered realistic.

TABLE 4.5A

FACTORS USED IN HEAT TRANSFER CALCULATIONS

Hot Channel Factors	
Fuel Content Per Plate Variation	1.02
Reactor Power Measurement Variation	1.05
Hydraulic Diameter Variation	1.05
Channel Cross Sectional Area Variation	1.10
Velocity Distribution	1.10
Product	1.36
Hot Spot Factors*	
Variation of Fuel Core Alloy Area	1.08
Fuel Content Per Plate Variation	1.02
Variation of Fuel Core Alloy Thickness	1.10
Reactor Power Measurement Variation	1.05
Correlation on Heat Transfer Coefficient	1.25
Velocity Distribution	1.09
Product	1.74
Power Distribution Factors	
Horizontal Power Distribution Max/Avg	1.87
Vertical Power Distribution Max/Avg	1.40
Fraction of Reactor Power Generated in Fuel Plates	0.90
Additive Inlet Water Temperature Variation	3°F

* Hot Spot is assumed to be 11 in. below core line.

Exceeding the saturation temperature by 20°F at the hot spot pressure does not appear to present undue hazard for the following reasons. Release of fission products into either the MIR or EIR primary coolant systems has never been related to overheating of a fuel plate. It is a known fact that the saturation temperature must be exceeded by some margin before nucleate boiling commences. The probability of all the factors cited in Table 4.5A occurring simultaneously is low.

No factors are included in Table 4.5A for uncertainties in the power distribution and maximum power density even though large uncertainties may exist in the physics calculations upon which these values are based. The specified operating velocity of 44 ft/sec rather than the 42.5 ft/sec obtained in Figure 8.7A allows for some additional uncertainty in the maximum power density. The recommended operating conditions for the reactor core are presented in Table 4.5B.

4.52 Primary Coolant System

4.521 General Description (Figure 4.5A)

Operating conditions of the primary coolant system are summarized in Table 4.5B. The system provides reactor cooling at a total flow of 30,000 gpm and a reactor inlet pressure of 300 psig. The four primary coolant pumps (7,500 gpm, 280 ft TDH, 700 hp) discharge water via the 36-in. reactor inlet header into the upper section of the reactor vessel. Water passes down through the core and out through four 16-in. outlet pipes which connect to the main 36-in. return header. Because it will probably be desirable to operate the four lobes of the reactor at different power levels to accommodate specific experimental requirements, the coolant flow through and from the reactor is divided into four parts to permit measurements of the operating power of each quadrant of this reactor. Separate flow and temperature measuring instrumentation is provided on each of the four outlets. The water from the four 16-in. outlet pipes flows into a common 36-in. return header, through four banks of two primary heat exchangers each, and then returns to the suction side of the primary pumps.

Two emergency pumps (2,000 gpm, 15 ft TDH, 20 hp), one of which is a spare, are connected in parallel with the main primary pumps to provide shutdown cooling. One pump is connected to the diesel power system to provide cooling in the event of commercial power failure. A firewater line is connected to the system to provide emergency cooling in case all other cooling systems fail.

The four primary pumps are centrifugal, controlled leakage pumps, each connected to a pair of heat exchangers. Heat exchanger characteristics are listed in Table 4.5C. Shell and tube heat exchangers with a floating tube sheet are indicated since the temperature differences involved are probably greater than can be

TABLE 4.5B

PRIMARY COOLANT OPERATING CONDITIONS

Reactor Power	250 Mw
Reactor Maximum Power Density	2.5 Mw/l
Reactor Inlet Temperature	130°F
Reactor Exit Temperature	187°F
Maximum Fuel Surface Temperature	420°F
Reactor Inlet Pressure	300 psig
Core Pressure Drop	92 psi
Average Coolant Velocity in Core	44 ft/sec
Core Flow Rate	25,400 gpm
Total Reactor Flow Rate	30,000 gpm
Number of Primary Coolant Pumps	4
Pump Horsepower	700 each
Number of Heat Exchangers	8
Total Heat Exchange Area	4.7×10^4 ft ²

TABLE 4.5C

HEAT EXCHANGER SPECIFICATIONS

Heat Transfer Area	5,870 ft ²
Tube Side Fluid	Primary Coolant
Shell Side Fluid	Secondary Coolant
Tube Length	21 ft
Tube Outside Diameter	5/8 in.
Tube Wall Thickness	16 BWG
Tube Configuration	Triangular
Tube Pitch	1-1/8 in.
Tube Side Pressure Drop	10 psi
Tube Material of Construction	304 stainless steel
Overall Shell Length	27 ft
Shell Outside Diameter	42 in.
Shell Thickness	1/4 in.
Baffle Type	Segmental
Baffle Spacing	72 in.
Shell Side Pressure Drop	5 psi
Shell Material of Construction	Mild Steel

sustained by fixed tube sheet construction. This choice probably increases the preventive maintenance necessary to insure that the tube sheets do not permit primary coolant to leak into the secondary coolant. However, preventive maintenance is much more easily accomplished in comparison to the repair of a cracked tube within a fixed tube sheet. The over-all heat transfer coefficient of these exchangers under operating conditions has been set at 400 Btu/hr, ft², °F; heat transfer rates of this magnitude have been obtained consistently during ETR operation and the coefficient has not shown a decrease as a function of operating time. The specifications for the heat exchangers in ATR are given in Table 4.5C.

The required coolant flow is controlled by the main butterfly valve, the position of which is set prior to startup. Since only steady flow conditions are required, only minor adjustments of this valve should be required during operation. Primary coolant flow is measured by a Gentile Tube in the reactor inlet line.

A 4,000 gal capacity surge tank is connected to the main coolant piping on the pump discharge to dampen pressure transients which could result due to rapid coolant temperature fluctuations.

The system is pressurized by means of two pressurizing and degassing pumps (300 gpm, 500 ft TDH, 50 hp), one of which is a spare. The pumps take suction from a 5,000 gal capacity degassing tank and discharge coolant into the main system near the suction side of the main coolant pumps. The flow from the two degassing and pressurizing pumps is measured by means of an orifice which, via a flow recorder-controller, positions a flow control valve to maintain a constant flow from the degassing system into the main coolant loop. Primary water is bled from the main loop system via a pressure control valve positioned by means of a pressure recorder-controller which senses the reactor coolant inlet pressure. The volume of coolant flowing through the back pressure control valve to the degassing tank is equal to the volume of liquid being pumped continuously into the main system by the pressurizing pump, so that a constant primary coolant pressure is maintained. The degassing tank, normally 1/2 full, is equipped with level controls, an overflow, and an air purge to maintain a low concentration of decomposition and fission gases above the water.

The reactor thermal shield is cooled by 200 gpm of primary coolant taken off the inlet header. The temperature of the coolant leaving the thermal shield is regulated by controlling the coolant flow.

Two gland seal and pressurized pumps (50 gpm, 600 ft TDH, 10 hp) provide demineralized water for the rod seals, experiment tube seals, primary pump glands and the HDW system. Only one pump is used normally with the other pump serving as a spare.

4.522 Primary Cooling Water Chemistry (Fig. 4.5B)

The presence of aluminum, beryllium, and stainless steel in the reactor requires that the pH and electrical resistivity of the light water coolant be rigidly controlled. Due to the possibility of a fuel element rupture, provisions must also be made for removing fission products from the coolant.

The purity of the water is controlled by the primary by-pass demineralizer which maintains the pH of the coolant between 5.5 and 6.5, and the specific resistivity at more than 250,000 ohms, in addition to removing fission products. Water is taken from the primary supply header, passed through the demineralizer system and then back into the primary discharge header at a rate of 500 gpm. The system consists of a prefilter and a postfilter, two cation, and two anion beds valved so that the water passes through either or both of the cation beds, and then through either or both anion beds. The filters and any one or all of the resin beds can be bypassed. The system is designed so that in the event of a fission break the primary coolant can be passed through the demineralizer system to the leaching pond. The concentration of fission products in the primary coolant can be reduced by a factor of 1,000 in approximately 14 hours with the 500 gpm primary coolant flow rate through the demineralizer.

Past experience on similar systems at the ETR and MTR has shown that the cation bed becomes too radioactive to allow resin regeneration. Therefore, the cation beds are designed so that the bed may be changed but not regenerated. Past experience with the ETR demineralizer indicates that it is permissible to regenerate the anion beds. Therefore, the anion beds are designed so that the resin can be regenerated or changed.

A degasifier is provided to remove gaseous fission products, oxygen, hydrogen, and nitrogen from the primary coolant system. Two 1,000 gpm demineralized water flush pumps are provided to purge the primary cooling water system after shutdown with demineralized water from the MTR 300,000 gal storage system.

4.53 Secondary Cooling System (Fig. 4.5C)

Heat generated in the reactor core, experiments, D₂O reflector system, and the borated H₂O control system is transported by the secondary coolant system from heat exchangers to the cooling tower where most of the heat is released by evaporation to the atmosphere. The total secondary coolant flow is 42,000 gpm including 6,000 gpm of utility cooling water. The secondary coolant on-tower and off-tower temperatures are 140°F and 85°F, respectively. Four vertical turbine pumps (9,000 gpm, 125 ft TDH, 350 hp) circulate the secondary water to the shell side of the primary heat exchangers,

D₂O heat exchangers, and the borated H₂O heat exchangers with flow regulated by temperature controlled butterfly valves. The total secondary flow, and the coolant flow and inlet and outlet temperatures to each of the system heat exchangers, are measured and recorded.

4.531 Cooling Tower

A forced-draft cooling tower consisting of eight cells is provided for final heat dissipation to the atmosphere. Each cell is 36 ft long, 36 ft wide and 46 ft high. Each cell contains a fan driven by a two-speed, 75 hp motor; fans are reversible to permit de-icing operations during the severe winter weather. The tower was selected on the basis of the following parameters:

42,000 gpm maximum flow

10⁹ Btu/hr heat load

65°F maximum wet bulb

85°F off-tower temperature

140°F on-tower temperature

20°F approach

4.532 Secondary Cooling Water Treatment (Fig. 4.5C)

Chemical treatment of the raw water fed to the cooling tower is necessary for four reasons:

1. to inhibit scale formation,
2. to minimize corrosion,
3. to discourage algae formation, and
4. to minimize the degeneration of the cooling tower redwood.

Three chemicals are injected:

1. 66°Baumé sulfuric acid,
2. gaseous chlorine, and
3. dianodic (a mixture of polyphosphate and dichromate).

The injection rates are given in Section 4.814, "Cooling Tower Make-up Water".

4.54 Pure D₂O Reflector System (Fig. 4.5D)

The D₂O reflector material is circulated through a tube-and-shell exchanger (cooled by secondary water) to remove the heat generated therein. The D₂O enters the upper reflector area and exits from the lower area via 6-in. stainless steel supply and return lines. Circulation is accomplished by sealed stainless steel pumps.

The coolant system is provided with by-pass facilities for deuterated-resin ion exchange columns for purification and pH adjustment. A positive helium atmosphere is maintained above the D₂O in the storage and surge tanks to prevent contamination by the moisture (light water) contained in the atmosphere. Also included are facilities to recombine D₂O which has been dissociated in the reactor reflector.

PPCO - C - 2725

FIG. 4.5 B
PRIMARY BY-PASS DEMINERALIZER
FLOW DIAGRAM

SECONDARY COOLANT SYSTEM - FLOW DIAGRAM

FIG. 4.5D
PURE D₂O REFLECTOR SYSTEM
FLOW DIAGRAM

4.6 Critical Facilities

4.61 Introduction

The performance of the ETR Critical Facility (ETRC) has aptly demonstrated that an exact nuclear mockup (referred to as a critical facility in this report) is an important adjunct to a high flux test reactor. Since its initial criticality date, May, 1957, the ETRC has been used constantly to support the ETR program.

Parameters which can be determined in a low power reactor to support the operation of a test reactor are: 1) neutron flux distributions, 2) gamma heat generation rates, 3) excess reactivities, and thus charge lifetimes, 4) shutdown reactivities, 5) rod, or control region, calibration, 6) consequences of experiment failures, 7) fuel loading requirements, reactivity and neutron flux, and 8) effects of the insertion or removal of experiments. Many of these measurements are essential to the operation of the reactor and if not made in a critical facility would have to be made in the power reactor itself. Others serve to aid in the design of the experiments and if not available would result in something short of an optimum design.

One of the most difficult reactor parameters to calculate accurately is the critical mass of operational fuel loadings. If a critical facility were built far enough in advance of the test reactor, the first core of fuel elements or prototype elements built could be used to determine any required adjustment in the fuel content of the elements. Another possibility considered for the proposed reactor is graduated fuel density within the element. The required non-uniform distribution would be determined from measurements in this preliminary loading. This could be accomplished by having a few special elements with removable plates so that plates with a variety of fuel densities could then be inserted.

At the same time the above determinations were being made, data would be obtained for designing the experiments. These data would be obtained by making neutron and gamma flux measurements in prototypes of the experiments.

Because no reactors have used the type reflector control proposed, probably less is known about this one factor than any other feature of this reactor. From measurements in a critical facility, it will be possible to determine the boron concentrations required to control the excess reactivity and the changes in the flux distributions due to the poisoned reflector.

4.62 Description

The total ground floor space which is recommended for this facility is 3000 sq ft in a room 50 ft x 60 ft. This includes space for a 10 ft x 24 ft x 20 ft deep canal for the facility, floor space

for experiment mockup assembly and storage, reactor console, and office space for the staff. This room should be located in the main reactor building and the canal should be connected to the transfer canal so that radioactive fuel and samples can be transferred from the reactor.

The reactor core should be an exact nuclear mockup of the core proposed. The structural requirements can be relaxed, of course, since the core will not have to withstand the hydraulic pressures created by coolant flow. Since the pressure vessel is close to the core, the nuclear equivalent of these must also be included. It is necessary that it extend only a few inches above and below the core, however.

The mechanical design of the ETR Critical Facility has been found to be quite adequate. Therefore, it is proposed that this facility be built similarly. Since a quite complete description appears in the hazards report¹, only a brief description will be included here.

The grid plate which supports the core, reflectors, and neutron detector tubes rests on the bottom of the canal. A control bridge which contains all rod drives rests on the canal parapet. The control wiring is fed through quick-disconnect plugs so that the control bridge can be removed to gain access to the core. The bridge is removed and placed in dry dock with an overhead crane. Although this has been quite satisfactory in the ETRC, an alternate method, which eliminates the use of the crane for this purpose, would be to mount the control bridge on wheels and tracks, and roll the bridge from above the core. The bridge would have to be moved toward the end of the canal so that the fuel could be moved freely from the transfer canal to the core. This would require a slightly longer canal but the additional length would have to be only deep enough to receive the control rod drives. Provisions would also have to be made for servicing the control rod lifting magnets.

A working bridge which permits working directly over the core is mounted on wheels and rests on the parapet.

The control and safety instrumentation of the ETRC is also offered as an example of a suitable system. The ETRC system has proved to be very reliable and easy to maintain. It is composed principally of commercial instruments, many of which are also used in the MTR and ETR. This is especially desirable because repair is easier since the instrument repairmen are familiar with the instruments and since additional spare parts are not required. Since the shim control for the proposed reactor is principally reflector control, pumps will have to be substituted for rod drives. This should not change the mode of operation, however.

1. D. R. deBoisblanc, "The Engineering Test Reactor Critical Facility Hazards Summary Report", IDO-16332, March 27, 1957.

The reflector control system should have the same range of reactivity control as does its parent reactor. Its rate of change would not have to be as fast, however, since xenon override is not necessary. To avoid unnecessary waste of experimental time, the response should not be unduly slow, however.

It is proposed that the borated H_2O flow system be similar to that in the parent reactor except that no heat exchangers will be required. The recirculating pump would serve only as a mixer and would operate whenever the reactor is operating.

Using the plant H_2O boron removal system looks attractive from an economical viewpoint and appears to be feasible but has one disadvantage. If the critical facility goes into operation long before the power reactor, the plant H_2O boron removal system may not be completed soon enough. Therefore, since a removal system of the capacity required for the critical facility is a small fraction of the cost of the entire facility, a separate system may be advisable. It is estimated that a system with a capacity of 100 gal per day will be adequate to serve this facility. This is based on the assumption that the system contains dump and holdup tanks so that it will not be necessary to remove the boron from the H_2O each time an excess reactivity or some other large reactivity measurement is made. Boron would have to be removed only when it becomes necessary to purify the H_2O .

4.7 Site, Buildings, and Facilities

4.71 Site Plan

The proposed plot plan for the ATR site is shown on Fig. 4.7A - Site Plan. ATR is located west of and adjacent to the north end of the present MTR-ETR site. The sharing of certain existing facilities, the availability of technical and operating personnel, and the existence of an adequate and established degree of separation from surrounding populace favorably affect the proposed location.

4.72 Reactor Building

Design philosophy leading to the layouts shown on Figures 4.2A, 4.2B, 4.2C, 4.2D, and 4.2E, have been discussed in Section 4.2 - Reactor Building Complex. Some of the more important details pertaining to this area are discussed in material following.

The building section housing the reactor, canal, critical facility, and operating office areas has been kept clean and unencumbered. Means are also provided for isolating any individual operating area from other areas when required.

The reactor building proper is of insulated panel construction with an overall main floor area of some 27,000 sq ft. Of this area approximately 10,000 sq ft are devoted to the reactor proper, 3,000 to the critical facility, and another 8,000 sq ft occupied by the main canal and its access and operating areas. The remainder of the first floor is utilized for health physics, decontamination room, tool room, offices and rest rooms. Heating and ventilating equipment is housed in a room located above the first floor offices in the southeast corner of the Reactor Building.

4.721 Reactor Area

The reactor area is 100 ft by 100 ft with a clear, unobstructed, minimum height of 45 ft. The working canal extends under the north wall of the area up to the north edge of the reactor structure. The reactor control room, shift supervisors' office, passenger and freight elevators, truck and personnel access doors, and an HP issue room are located along the south wall of the reactor area. This area is serviced by a large 50 ton bridge crane and by a 2 ton handling crane.

The first basement level and the second basement level of the reactor building are devoted to experiment cubicles, control panels, reactor instrument room, reactor instrument repair room, switch gear, a fuel storage vault, a counting room, data reduction facilities, etc. as described in Section 4.2. Numerous hatchways, stairwells, and elevators provide communication between the three floors for equipment and personnel.

4.722 Canal Area

The reactor is connected to the main canal by means of a working canal lateral. In addition to the critical facility, an experimenters' canal and work area are connected to the main canal. The main canal will be some 20 ft in depth, 8 ft wide and 176 ft long. The experimenters' canal, located on the west end of the main canal, will be 20 ft deep, 8 ft wide and approximately 15 ft long. This canal, although a part of the main reactor canal, will be equipped with a bulkhead to separate it when required. A similar bulkhead will be provided for the critical facility.

Working areas are provided on both sides of the main canal. A through truck aisle parallels the main canal. This whole area is serviced by a 50 ton bridge crane which permits convenient loading and shipping of experiment casks.

It is anticipated that the canals will be lined with stainless steel to minimize the leak potential and to keep canal water contamination at the lowest possible levels. Demineralized water at a pH of 6.5 will allow storage of fuel elements, experiments, etc., for long periods of time with little or no corrosive attack.

4.73 Process Water Building

The process water building is situated immediately adjacent to and north of the reactor building, with an enclosed motor area separating the two buildings. This motor area is serviced with an 8 ton hoist. The building proper is heavily shielded for protection against a potential gross fission break into the reactor cooling water stream. The main coolant pumps are located at approximately grade elevation and are connected to drive motors by shaft extensions through the shielding wall. A heat exchanger pit 24 ft wide by 60 ft long by 30 ft deep is provided to accommodate ten vertical heat exchangers and the necessary piping. An overhead 10 ton crane serves the heat exchanger and pump areas and has a lift adequate to pull any exchanger from its pit to the grade level floor for maintenance work.

A control room for the reactor cooling system is provided adjacent to the utility building.

4.74 Utility Building

The utility building is situated immediately adjacent to and east of the process water building, and is constructed of pumice block or equivalent. Two heavy duty diesel generators, switch gear, control panels, etc., are located on the first floor. This area is serviced with a 10 ton crane. Directly below this area is the high pressure demineralized water equipment area. The emergency cooling loop (ECL) equipment is located adjacent to the high pressure demineralized water equipment area.

4.75 Laboratory and Engineering Building

A one-story pumice block or equivalent building with approximately 5,000 sq ft of usable area is contemplated to house the laboratory, engineering, and technical department personnel necessary to support the reactor and experimental program.

4.8 Utilities

The ATR utilities are combined with the existing MTR-ETR utility systems in those cases where the existing utilities are adequate or can be expanded to provide sufficient capacities for the three reactors. By providing common utilities, substantial savings are realized both in construction and operating costs.

4.81 Water

4.811 Wells and Storage (Fig. 4.8A)

One additional well (3,000 gpm, 600 ft TDH, 600 hp) is provided to supplement the three existing MTR-ETR wells which have a combined capacity of approximately 7,700 gpm. Any two of the three large wells are sufficient to supply the water requirements of the three reactors operating at design power.

The three existing raw water storage tanks with a total capacity of 1,500,000 gal store sufficient water to satisfy the fire requirements of the three plants since simultaneous fires in two or more cooling towers is considered to be remote.

4.812 Firewater

The existing two MTR-ETR firewater pumps provide adequate firewater for the ATR area. Both of the pumps are rated at 5,100 gpm at 200 ft TDH. The normal pump is driven by a 300 hp electric motor and the stand-by pump is driven by a 490 hp gasoline engine.

The ATR area is looped with a 12-in. firewater main connected to the existing MTR-ETR firewater loop. Fire hydrants with a maximum spacing of 300 ft are placed at strategic locations around the plant and each building is serviced by at least two hydrants. Hose stations are provided in all buildings. The cooling tower is protected by a fire detection system which automatically opens the deluge valves to the cooling tower bay or bays involved in a fire and also actuates the deluge valves on the bays adjacent to the fire. The maximum number of tower bays that will have to be flooded at any one time is six. At 0.5 gpm per sq ft of deck area this is about 3,900 gpm. In addition, about 500 gpm of firewater is required for spraying six fan motors above the deck and 600 gpm for one fire hydrant. Based on these estimates the existing firewater pumps, rated at 5,100 gpm, are considered adequate.

An interconnection to the primary coolant system is provided to supply raw water to the reactor in case of an incident involving a large loss of primary water.

4.813 Potable Water

The MTR potable water system has a design rating of 100 gpm. Potable water usage has averaged about 30 gpm during the last six months for the ETR and MTR facilities so that the additional requirements for ATR can easily be met from the existing system. The potable water is transferred from the MTR demineralizer building to the ATR area via a 4-in. underground carbon steel line. The existing potable storage water tank has a capacity of 10,000 gal and the two potable water pumps each has a capacity of 100 gpm. The total hardness of the potable water is adjusted to about 50 ppm by blending raw water and softened water. The water is chlorinated and is intended for sanitary purposes only.

4.814 Cooling Tower Make-up Water (Fig. 4.8A)

Either one of two cooling tower make-up pumps (2,500 gpm, 100 ft TDH, 100 hp) supplies the evaporation, windage, and blowdown water requirements of the cooling tower. The pumps are located in the MTR raw water pump house extension and furnish make-up water to the cooling tower through a 12-in. carbon steel line. The cooling tower make-up requirements are based on maintaining four cycles of concentration* in the cooling tower water. MTR-ETR experience has indicated that seven cycles of concentration are satisfactory at a water temperature of 110°F. However, the ATR secondary on-tower water temperature is 140°F and it is deemed advisable to limit the maximum cycles of concentration to four, thereby minimizing scale formation in the heat exchangers.

Make-up water treatment is required to inhibit corrosion, scaling, and the growth of slime and algae. When the reactor is operating at full power, it is estimated that 350 gal of 66° Be sulfuric acid, 300 lb of Dianodic (a corrosion and scale inhibiting chemical), and 40 lb of chlorine are required per day.

4.815 Demineralized Water

The MTR demineralized water production during the last six months has averaged less than 150 gpm for the MTR and ETR facilities. The ATR requirements are estimated to be 150 gpm or less, so that the present MTR production facilities are probably adequate.

An additional 100,000 gal stainless steel demineralized water storage tank is provided for ATR increasing the total maximum storage capacity to 300,000 gal. The new storage tank is connected in parallel to the two MTR 100,000 gal tanks.

* Cycles of concentration is the term employed to indicate the degree of concentration of the circulating water as compared to the make-up water.

Two new demineralized water pumps (300 gpm, 175 ft TDH, 30 hp) are provided to replace the existing pumps so that either pump has sufficient capacity to furnish the anticipated demands. The pump motors are connected to the MTR emergency diesel-driven electrical power so that demineralized water is available during commercial electrical outages.

The existing two MTR flush pumps (1,000 gpm, 174 ft TDH, 60 hp) are adequate to serve the three reactors since they are used for flushing the reactor primary systems prior to removal of the top reactor vessel closure and as stand-by for the 300 gpm demineralized water pumps during periods of peak demand. The probability of being required to flush the three reactors simultaneously is considered remote and would probably only occur after a commercial electrical power outage at which time the demineralized water flush for one or more reactors might be delayed; however, with 300,000 gal storage capacity provided, sufficient water should always be available for flushing the three reactors.

Demineralized water from the MTR demineralizer building is pumped to the ATR area via an 8-in. stainless steel line and distributed throughout the reactor, process, and utility areas for make-up water to the experiments, the gland seals, the primary system degassing tank, the primary pump seals, etc.

4.816 Water Treatment Chemical Storage

Sulfuric acid is stored in the existing two 10,000 gal MTR acid storage tanks. MTR-ETR usage for cooling tower make-up treatment averages less than 1,000 gal per week and the increased usage for the demineralizer at a net water production rate of 300 gpm amounts to approximately 1,000 gal per week. The maximum estimated acid requirements for the ATR cooling tower water treatment is 2,000 gal per week, so that the existing storage provides a four to five week supply. Acid delivery can be made at the site within three to four days after an order is placed. Under these circumstances, the existing MTR acid storage capacity is more than adequate.

Two 12,000 gal tanks provide storage for 306,000 lb of 50 per cent solution caustic soda. With a net demineralized water production of 300 gpm, the caustic demand is about 1,100 gal per week, so that the existing storage provides over 20 weeks' supply which is considered to be more than adequate.

Both caustic and acid truck unloading facilities are adequate since a truck can be unloaded in one to two hours. The MTR acid and caustic transfer pumps are adequately sized to handle the increased acid and caustic requirements.

Dianodic is purchased in 400 lb fiber drums and is stored in the MTR or CF warehouse. A week to ten days' supply of this

material is maintained in the secondary system pump house for immediate use. Chlorine is purchased in 150 lb cylinders and stored in the warehouse with several cylinders stored in the secondary pump house for immediate use.

4.817 Utility and Emergency Cooling Water (Fig. 4.8B)

The UCW (utility and emergency cooling) system is used as the heat transfer medium for final heat removal by the cooling tower on the reactor auxiliary systems. These include the high pressure demineralizer water (2,000 gpm), emergency cooling loop (425 gpm), reactor canal cooling (700 gpm), diesel engine cooling (1,200 gpm), and emergency cooling for the primary heat exchangers in the event of a commercial power outage (1,500 gpm).

Two motor-driven, vertical turbine pumps (3,000 gpm, 150 ft TDH, 150 hp) are located in the cooling tower pump house and provide the necessary water flow for the system at the design off-tower temperature of 85°F. In the event of a commercial power failure, one UCW pump will continue to operate, providing cooling tower water to critical cooling loads. It is anticipated that the total heat load imposed on the cooling towers by the utility cooling system will be 20 to 25 Mw.

4.818 Experimental Emergency Cooling Loop (Fig. 4.8C)

The ECL (experimental emergency cooling loop) is a system which provides cooling of in-pile tubes and/or fuel elements during periods when the experimenters' coolant circulating system is out of service.

The shielded loop is designed to cool defected specimens under emergency conditions. Therefore, the entire loop system is constructed of stainless steel, which will permit use of various chemical decontamination solutions.

The ECL system circulates high purity demineralized water up to 500 gpm at a pressure of 450 psig and a temperature of 150°F. The experimental loops return the water at 200 psig with temperatures up to 190°F thus forming a closed loop. The heat load of 3 Mw is transferred to the utility cooling water system via shell and tube heat exchangers for final dissipation through the cooling tower.

System pressure is maintained at 450 psig by means of a pressurizing pump taking suction from the low pressure demineralized water system and discharging through a pressure regulating valve to the circulating pump suction headers. To compensate for system pressure and volume changes during temperature transients, a surge tank with a pressurized nitrogen blanket is installed on the circulating pump discharge header.

A mixed bed ion exchanger column is provided to control cooling water pH and conductivity and to maintain the system demineralized water purity. A removable shielded resin cask is provided to dispose of depleted ion exchange resins. The loop instrumentation is located on a control panel outside the loop cubicle shield.

4.819 High Pressure Demineralized Water (Fig. 4.8D)

The HDW (high pressure demineralizer water) system at a supply pressure of 300 psig circulates 3,000 gpm of demineralized water of which 2,500 gpm is supplied at 150°F and 500 gpm at 100°F. The 2,500 gpm system removes fission and gamma heat from the experimental loop circulating water via water-to-water experimental heat exchangers. The 500 gpm system, at 100°F, is used for experimental primary pump and ion exchange cooling, both of which require lower temperature cooling water.

The maximum heat input to this system is estimated to be 16 Mw which is transferred through shell and tube heat exchangers to the utility cooling water for heat removal by the cooling tower. Suitable temperature indicators and flow measuring devices are provided on both the demineralized water side and the utility water side of the heat exchanger to control the system temperature.

The three HDW circulating pumps are horizontal centrifugal, double suction, double volute type with a capacity of 1,000 gpm each at 230 ft TDH. The pumps are driven by 100 hp electric motors. A by-pass pneumatic pressure regulator valve is installed between the supply header and the common return header to maintain a constant supply pressure under varying experimental demands. In the event of a commercial power outage, one HDW circulating pump continues to operate providing water at reduced flows to the experimental loops for shutdown heat removal. This necessitates the experimental uses to be automatically reduced to insure adequate cooling water for critical demands.

Demineralizer water from the gland seal and pressurizing pumps passes through a pressure reducing valve set at 200 psig to control pressure on the suction side of the HDW circulating water pumps. A system fill line by-passing the pressure regulating valve is incorporated to provide fast filling of the entire system.

4.82 Heating and Ventilating

4.821 Steam and Condensate

Steam at 135 psig for heating the various ATR buildings is supplied from the existing steam plant via a 4-in. insulated steam header. Condensate collected in the reactor

building condensate receiver (250 gal) is returned to the MTR steam plant by means of one of two condensate pumps (25 gpm, 100 ft TDH, 2 hp).

The installation of an additional 17,500 lb/hr boiler provides additional capacity to the three MTR steam generating units so that one of the four boilers can serve as a stand-by unit. Estimated maximum steam requirements for MTR, ETR, and ATR are 40,000 to 45,000 lb/hr, so that three boilers with a combined capacity of 52,500 lb/hr are more than adequate.

The boiler feed pumps, one steam turbine-driven (25 hp), and the other electric-driven (20 hp), are both rated at 90 gpm at 425 ft head. Either pump has sufficient capacity and head to deliver 45,000 lb of feed water per hour which is considered adequate. The two existing MTR steam plant condensate pumps (5 hp) are rated at 90 gpm at 75 ft head and are adequate for the estimated condensate load.

The MTR fuel oil storage, consisting of two 200,000 gal tanks, is sufficient capacity for more than three weeks operation at a continuous steaming rate of 45,000 lb and is more than adequate. Either of the two fuel oil pumps rated at 10.5 gpm is capable of supplying enough burner oil to the four 17,500 lb/hr boilers when all units are firing at their maximum rate. The fuel oil truck unloading pump with an original design rating of 50 gpm is sub-standard but this item is scheduled for replacement with a 100 gpm pump before ATR is completed.

4.822 Reactor Building

Reactor Operating Area. This area is provided with a separate heating and ventilating system furnishing air at 70°F by means of a supply duct system discharging air to the first floor area. Approximately 90 per cent of the air will exhaust through the experimental cubicles to dissipate the anticipated 75 kw experimental heat load per cubicle. The remaining air exhausts through the sub-pile and control rod access rooms. Pressures in these areas are below those of all adjoining areas so that air flow is from normal personnel working areas into cubicles which are potential areas of high air activity. All air from the cubicles discharges into the cubicle exhaust header and is exhausted through the stack. The heating and ventilating equipment for this area is located in the reactor building heating and ventilating room and consists of adjustable louvers, non-freeze preheat steam coils, a capillary washer filter system and steam reheat coils. A supply fan of approximately 66,000 cfm is required which provides six air changes per hour, five of which are outside air.

Offices and Hallways. These areas are provided with a separate heating system furnishing air at a minimum of 70°F. Supply ducts are routed to each office, with return air ducts located in the hallways. The heating equipment for these areas is located

in the heating and ventilating room and consists of adjustable outside and return air louvers, non-freeze heating coils, washable high efficiency filters, and mixing dampers. A blower of 1,500 cfm provides six air changes per hour, three of which are outside air.

Storage Canal Area. This area is supplied with an additional heating system furnishing air at a minimum of 70°F by means of supply ducts located on the first floor. The heating equipment is located in the heating and ventilating room and is similar to that described for the offices. A fan of approximately 25,000 cfm provides four air changes per hour, two of which are outside air.

Critical Facilities Area, Health Physics, and Decontamination Rooms. These areas are provided with a separate heating system furnishing air at a minimum of 70°F. Supply ducts are routed to each room and area with return ducts suitably located to provide a closed system independent of the other reactor facilities. The heating equipment is located in the heating and ventilating room and includes a supply fan of approximately 11,000 cfm which provides four air changes per hour, two of which are outside air.

First and Second Reactor Building Basements. The first basement is supplied with 19,000 cfm of air at 70°F from a separate heating system. Half of the air is returned to the heating equipment located in the heating and ventilating room while the other half discharges to the second basement where it is exhausted to the cubicle exhaust system via the sub-pile room, rod access room, tank vaults, cubicles, etc.

Heating and Ventilating Controls. Controls for each system originate downstream of the heating coils. They consist of thermocouple-equipped temperature recorder controllers and actuating pneumatic type modulating steam control valves to the heating coils. The reactor operating area system has an additional temperature recorder controller for the preheat steam coils.

Reactor Control and Equipment Rooms. These rooms are air-conditioned by means of a multiple package air conditioning unit. Fresh supply of air to this unit is from the reactor area supply system. Design conditions are 70°F $\pm$ 1°C and 40 per cent relative humidity for both summer and winter operation. Automatic temperature control is provided to adjust heating and cooling fluctuations. The air conditioning unit is located in the heating and ventilating equipment room. Total air circulation is six air changes per hour, two of which are outside air.

4.823 Supporting Facilities

Process Water Building. This area is heated to 60°F by means of industrial unit heaters. This area is exhausted through the cubicle exhaust system in order to maintain the area at

a negative pressure. Air activity problems are thus minimized in the adjoining operating areas.

Utilities Building. The utilities building houses the electrical switchgear, the motor-generator sets, the battery banks, and the two diesel generators. Since it is generally a machinery area, it is heated to 60°F by means of industrial type steam unit heaters equipped with fans, dry type washable filters, non-freeze heating coils, and dampers. Fresh air intake is two air changes per hour with exhaust through power roof ventilators.

Summer ventilation required to dissipate heat from the switchgear, M-G sets, diesel engines, etc., will be supplied by evaporative coolers installed on the intake side of the unit heaters. Maximum ambient room temperature is held to 40°C (105°F).

System control is from a single control center. It consists of thermostats actuating modulating control valves on the steam and evaporative cooling systems, and start-stop controls for the unit heater fans and roof ventilators.

Secondary Pump House. This building, which does not require continuous occupancy, is heated to 60°F by means of industrial steam unit heaters. Ventilation is provided by gravity roof ventilators.

Laboratory and Engineering Building. A central heating system furnishing air at a minimum of 70°F is installed in this building. The unit consists of a fan section, filter box equipped with dry type washable filters, heating coils, and a damper section. It is furnished complete with automatic controls to provide heat and ventilation to the office areas. Supply and return duct systems are provided with a total air circulation of six air changes per hour, two changes being outside air. Change rooms and toilets are exhausted to the atmosphere by separate exhaust fans at twelve air changes per hour.

4.83 Plant and Instrument Air System (Fig. 4.8E)

The MIR-ETR plant instrument air compressors and auxiliary equipment are located in the MIR steam plant. There are four compressors, three 300 scfm compressors and one 200 scfm stand-by compressor. The air compressors are a single horizontal tandem, double-acting, two-stage water-cooled compressors with three-stage step control. Features include automatic loading and unloading, spring backed graphite type carbon rings, non-lubricated packing, and dry type air filters. The discharge from the compressors passes through three 300 scfm dual absorption type air driers. These desiccant type driers are complete with steam regeneration systems and are equipped for semi-automatic operation.

Two large air surge tanks receive the moisture free air and contain sufficient holdup to provide a flow of 450 scfm for ten minutes with the pressure ranging between 165 to 115 psig. Control valve stations reduce the pressure for the plant air system to 150 psig and for the instrument air system to 115 psig. A third control valve shuts off the plant air distribution system if the pressure in the surge tanks drops to 115 psig. This maintains a continuous instrument air supply in the event of an unusually high plant air demand.

The existing MTR-ETR plant and instrument air system is supplemented by one 600 scfm, 165 psig, 150 hp motor-driven air compressor discharging to the supply manifold on the three original 300 scfm dual absorption type air driers. A fourth 300 scfm desiccant type drier connected in parallel to the existing drying facilities provides a total 1,200 scfm capacity for the overall MTR-ETR and ATR complex. One of the existing 300 scfm motor-driven compressors is converted to an emergency unit by connection to the ATR diesel generated power system. The converted 300 scfm emergency compressor supplements the 200 scfm emergency compressor energized by the MTR diesel generated emergency power system.

The 3-in. plant and instrument underground air lines servicing ATR are connected to the respective headers downstream of the system control valves located at the MTR steam plant. The large 600 scfm motor-driven compressor is located in the maintenance and storage area of the MTR steam plant with the necessary cooling water and control air supplies extended to the new unit.

4.84 Waste Disposal

4.841 Liquid Waste

Cold Waste (Fig. 4.8F). Non-radioactive liquids from the cooling tower and the utilities building flow directly by gravity to the plant sump. The warm plant effluents also discharge to the plant sump and are diluted with the cold wastes before being discharged to the leaching pond. Two motor-driven plant sump pumps (2,000 gpm, 25 ft TDH, 20 hp) pump the liquid wastes directly to the leaching pond via a 10-in. concrete line for disposal to the subsoil by percolation and to the atmosphere by evaporation.

Warm Waste (Fig. 4.8F). Radioactive liquid wastes from the canal, the reactor building, the heat exchanger building, and the primary coolant system degassing tank flow by gravity through 2-in. and 4-in. stainless steel piping to the 5,000 gal stainless steel warm sump tank located in a shielded cubicle below the reactor building basement floor. Two motor-driven turbine pumps (200 gpm, 100 ft TDH, 7.5 hp) pump the warm wastes to the 10-in. stainless steel primary system flush line, which discharges to the plant sump. The warm sump tank is equipped with a level indicator controller which automatically starts and stops the warm sump pumps.

Hot Experimental Waste (Fig. 4.8G). Coolant and decontamination solutions from the experimental loops and the laboratories will be the only sources of wastes whose radioactivity concentrations are too high to permit ultimate discharge to the leaching pond. The hot wastes from the laboratories are estimated to be only a few gallons per week.

If experimental capsules or reactor fuel element failures contaminate the primary systems to such a degree that the primary coolant cannot be flushed directly to the leaching pond via the plant sump, the reactor will be shut down and the primary coolant circulated and decontaminated by means of the primary bypass demineralizer until the radioactivity concentration of the primary coolant is reduced to a low enough level to be discharged to the leaching pond.

Hot wastes from experimental loops will drain via 2-in. stainless steel lines to the 1,000 gal hot experimental waste sump tank located in a shielded cubicle below the reactor building basement floor. Either of two motor-driven pumps (50 gpm, 100 ft TDH, 2 hp) transfers the liquid via a 3-in. stainless steel line from the sump tank to the MTR hot waste storage tanks in the yard south of the MTR process water building.

The existing MTR tank truck loading facility consisting of a concrete shielding canopy and fill and vent piping is located immediately east of the hot waste storage area. A truck trailer with a concrete shielded 1,000 gal capacity transfer tank is positioned under the canopy during the hot waste loading operation. A 1 1/2-in. vent line and the 1 1/2-in. fill line are connected manually to the portable tank through quick disconnect couplings prior to the loading operation when radiation fields are low. When the portable tank is full as indicated through the 1 1/2-in. fill line totalizer-indicator recorder located in the instrument cubicle above the hot waste storage pump house, the lines to the portable tank are disconnected and the tank is then transported to CPP for unloading. The MTR hot waste storage and tank loading facility are considered adequate to handle the increased load from ATR experiments.

Sanitary Waste. The sanitary wastes from the ATR area drain by gravity from the various facilities via an 8-in. concrete sewer main which connects to the original MTR sewer system west of building MTR 614. The original MTR sewage treatment plant is adequate except that leaching facilities will probably require expansion.

4.842 Gaseous Waste (Fig. 4.8H)

The gaseous waste disposal system exhausts air from potentially high air activity areas such as the experimental cubicles, the primary pipe tunnel, the process water building, and

the sub-pile room. It also purges the hot and warm waste tanks and the primary system degassing tank.

Provisions are made to exhaust the reactor working platform when the reactor vessel top closure is removed for in-tank work, so that gaseous activity in the reactor vessel is exhausted to the cubicle exhaust system and does not contaminate the building air. A portable canal exhaust hood removes radioactive gases from the surface of the canal water when ruptured capsules are stored in the canal below this hood.

Total waste gas quantity, which will be monitored prior to disposal through a 150 ft stack, is estimated at 70,000 cfm. The gaseous and particulate monitor is described in further detail under "Radiation Monitoring" of this report.

Two blowers (35,000 cfm, 25 hp) are provided and operated in parallel. One blower motor is connected to the diesel power bus and the other to the commercial power bus. System flow and pressure are recorded in the primary control room.

Over-pressure safety devices (rupture discs and pressure relief valves) on the experimental loops discharge into an experimental vent header which discharges into the cubicle exhaust duct on the downstream side of the blowers. Steam and radioactive gases when released from an experimental loop are thus vented to the stack to eliminate contamination of the experimental cubicles and equipment.

4.843 Solid Waste

The existing NRTS burial ground is considered adequate for the additional solid wastes from the ATR facility. The necessary hauling and burial equipment is also considered adequate.

4.85 Electric Power (Fig. 4.8I and 4.8J)

The electrical system requires two sources of power; one is commercial power supplied from the existing 132 kv power system and the other is diesel power supplied by diesel-driven electric generators installed at the ATR.

4.851 Commercial Power

The existing ETR 15,000 kva transformers are utilized to supply ATR electrical power at 13.8 kv. The installation of forced air cooling increases the rating of the two ETR transformers to 20,000 kva each, and since the ETR connected load is only about 11,000 kva per transformer, the transformers with the increased capacity are considered adequate to supply the additional load imposed by the ATR facility.

The ATR 13.8 kv feeders are connected to the ETR transformer secondaries and are routed through underground ducts to the ATR transformer yard where two 7,500 kva 13.8/4.16 kv transformers reduce the voltage to 4,160 volts. Electrical power at 4,160 volts is supplied to the large motors of 150 hp or more. Electrical power at 4,160 volts is further reduced to 480 volts for experimental and plant uses. Power is supplied for lighting, instrumentation, and fractional horsepower motors at 110 volts.

4.852 Diesel Power

Diesel power at 4,160 volts is supplied by two heavy duty diesel 1,500 kw generators, one of which serves as a stand-by unit. The critical plant and experimental loads are split between independent commercial and diesel power sources, i.e., a minimum of one pump on commercial and one pump on diesel power for any one system. When a commercial power failure occurs, the operating diesel generator continues to supply power to its normal connected loads. The reactor is automatically shut down on loss of commercial power or if the diesel generator fails. It is very unlikely that diesel generator failure and a commercial power outage would occur at the same time.

4.853 General

A tunnel with lay-in trays is provided for electrical cables from the utilities area to the process and reactor areas. A system of underground ducts and manholes is used for routing electrical power, communications, and alarm circuits to the other site buildings.

An additional 100,000 gal ground level, diesel oil storage tank adjacent to the existing MTR tank farm is provided to fuel the ATR diesel generators. It is estimated that 3,000 gal per day of diesel oil will be required when the diesel generator is running. The two MTR diesel oil transfer pumps each rated at 17.5 gpm at 35 psig are considered adequate since the peak consumption rate for all MTR-ETR uses is 7 gpm. The new diesel oil unloading pump, rated at 100 gpm, replaces the original MTR unloading pump (50 gpm) to provide more rapid unloading of diesel oil trucks.

4.86 Hot Cell

The existing MTR hot cell facilities are adequate for the additional work imposed by the ATR reactor and its experiments. However, additional manpower for hot cell operation is required.

Assembly and disassembly of irradiated NRB experiments is conducted at ECF as is presently done with NRB experiments from MTR and ETR.

4.87 Radioactive Monitoring

4.871 Direct Radiation

Area monitoring is performed in various parts of the plant with ion chambers incorporating built-in calibration sources. These are adjustable from the main control panel where the contact meter relays are located. All units alarm visually and audibly both at the detector and in the Health Physics office.

There are eight ion chambers on the first floor, seven on the first basement level and six on the second basement level. The control panels are in the Health Physics office and two multipoint recorders for recording levels of all 21 stations are located in the control room.

Ten ion chambers are located in the process water and utilities buildings. These alarm, as stated above, but do not record. Control units are mounted in the process control room.

4.872 Air

Constant air monitoring is accomplished by mobile scintillation units. One each in the utilities building and secondary pump house and two in the heat exchanger building record and alarm locally and alarm in the Health Physics office. Seventeen of the same units are located throughout the reactor building and record in the control room.

4.873 Fission Break

There is one primary coolant fission break detector. The process water sample is continuously drawn through a pressure regulator and flow meter to the monitor which may be either an iodine or delayed neutron monitor. The activity level is continuously recorded in the reactor control room.

4.874 Waste Gas

The waste gas monitor measures and totalizes air flow to the stack. It also monitors a continuous sample of the air stream for particulate and gaseous activity with a moving filter tape system. Scintillation counters with totalizers are used for both gas and particulate monitors. The instantaneous level of each records in the Health Physics office on a multipoint recorder.

4.875 Effluent Water

Effluent water to the leaching pond is continuously measured for flow rate and activity. A sample whose flow is proportional to discharge rate is monitored by a scintillation system.

Flow and activity are continuously multiplied and integrated to give the total curies discharged.

4.876 Personnel

Two hand-foot counters are located in the reactor building. Friskers of the quintector type are located at all reactor building exits, at elevator entrances, and at each stair well.

4.88 Communications

The communications system consists of two main parts, the page-intercom and the plant telephone net. These two systems supplement the commercially installed and serviced telephone system.

In addition, a manual, automatic coding fire alarm system is installed throughout the plant. This consists of alarm boxes and bells connected to a centralized relay and terminal strip unit. Present MTR-ETR alarms on firewater pressure, fire pump starting, and raw water storage will code the ATR alarms also.

This page-intercom system consists of a master paging network which utilizes many small speakers in order to adequately cover all areas in the reactor building. Page may be initiated from any of the master intercom stations. Master intercom stations are located at all important work areas. These stations are speaker-microphone units which connect to a central patch panel as well as to the paging system. Any two or more stations may be patched together allowing several networks in operation at any one time.

The telephone network consists of four telephone circuits with plug-in outlets for each circuit at all intercom locations as well as at all remote locations where instrument calibration may sometime be required. These four circuits terminate in the central patch panel so that any damaged circuit may be isolated and so that any of the four circuits may be patched together. This telephone system operates from a common dc source and uses carbon elements rather than sound-powered elements.

FIG.4.8A
RAW WATER SYSTEM - FLOW DIAGRAM

FIG. 4.8B
UTILITY COOLING WATER SYSTEM
FLOW DIAGRAM

FIG. 4.8C
EXPERIMENTAL EMERGENCY COOLING LOOP
FLOW DIAGRAM

FIG. 4.8 E
PLANT AND INSTRUMENT AIR SYSTEM - FLOW DIAGRAM

FIG. 4.8F
LIQUID WASTE DISPOSAL SYSTEM
FLOW DIAGRAM

P.P.CO.-C-3034

FIG.4.8G
EXPERIMENTAL HOT WASTE DISPOSAL SYSTEM
FLOW DIAGRAM

P. P. Co. - C-3036

FIG. 4.8H
GASEOUS WASTE DISPOSAL SYSTEM
FLOW DIAGRAM

FIG. 4.8J
DIESEL GENERATORS
PROCESS FLOW DIAGRAM

P.P.CO.-C-3038

5.0 EXPERIMENTAL FACILITIES AND SERVICES

The reactor physics, mechanical layouts, supporting facilities and utilities are discussed in Sections 3.0 and 4.0. The purpose of this section is basically twofold:

1. to discuss the factors which led to the design criteria used in the other sections;
2. to describe the types of experiments which determined the design of the reactor.

5.1 Irradiation Facilities

As in all test reactors, samples are irradiated in this reactor in two different manners. The samples may be "capsules" which make use of the reactor process cooling water for heat removal or may consist of loop specimens which are essentially capsules irradiated in the high pressure water loops. One can refer to these two classes of experiments as capsules and "loop specimens", respectively, even though in some cases the capsule type may bear little resemblance to the simple sealed container which the word capsule brings to mind.

5.11 The Reactor Physics Model

In test reactor evaluation, it is extremely critical that the physics model be properly specified. This is true because of the nuclear coupling between the experiments and the test reactor itself and because the radiation characteristics of a test region are highly dependent upon the loading of the test region. For example, the neutron fluxes in a dummy filler piece in the ETR core bear little resemblance to the fluxes in typical experimental systems placed in the same hole. Physics calculations for this reactor have been made on the basis of typical high pressure water loops in the experimental holes. These loops consist of stainless steel tubes containing fuel samples. The quantity of fuel selected and the flux levels supplied are based on modern high flux testing requirements. It should be noted that both the thermal and the "fast" flux available are dependent on the annular filler media and that a variety of absolute flux levels and spectral combinations may be obtained for any particular experimental configuration by varying the annulus composition.

5.12 Capsule-Type Experiments

The concept of a capsule as used in this design is a very general one. The capsules may vary in length from a very short length to full core length; they may consist of "baskets of slugs"; they may or may not have leads (if so, the leads are visualized as MTR-ETR type leads); sample diameter may vary up to 1.5 in. A capsule consisting of a number of instrumented subassemblies which must be removed from the reactor with leads intact for inspection or reassembly has also

been visualized, and provision has been made for removal of lead-type capsules and reinsertion with leads intact.

Leads are carried out through tank access nozzles to a capsule instrument panel in the second basement. Information from the capsules may also be processed in the data reduction system.

5.13 Loop Experiments - In-Pile Sections and Sub-Pile Room

The basic loop in-pile section around which this reactor is designed is a unit re-entrant from the reactor bottom head with reactor top cap penetration for experimental access (Fig. 5.1B). The tubes are insulated from the reactor water by conventional stagnant gas thermal barriers. Provision is made for shielded traps in the sub-pile room directly below the test sections. Allowance has also been made for a minimum of 1 in. of lead shielding around the experimental piping.

The test regions through the reactor core are sufficiently large to permit installation of in-pile tubes of this type having inside dimensions up to at least 2 in. in diameter. The test regions are isolated hydraulically from the reactor fuel and outside dimensions of the in-pile tubes need not meet rigid dimensional specifications. The material of construction considered for the in-pile tubes is Type 347 stainless steel. No prohibitive problems should be encountered due to gamma heat which is of the order of 20 watts/g.

The top closure plug of the loop is outside the reactor pressure vessel and reactor refueling operations and loop sample handling are relatively isolated from one another. Loop branch piping is carried from the in-pile tube to a piping corridor which surrounds the reactor.

Although the basic in-pile tube design is a unit re-entrant from the bottom, the reactor design includes access nozzles in the top of the pressure vessel and the pipe corridor also serves this region. This feature makes possible the design of experimental tubes re-entrant from the top and through loops which enter from the sub-pile room and exit from the upper nozzles or vice-versa.

5.14 Loop Piping Corridor

Around the reactor is a shielded loop piping corridor. In this area, loop piping may be manifolded, ECL tie-ins may be made, etc. This area provides isolation of a primary loop cubicle from the radioactively hot lines which exist when a loop's in-pile section is being operated on emergency cooling water.

5.15 The Loop Flow Diagram

The loop flow diagram, around which the reactor building and supporting facilities are designed, is shown in Fig. 5.1A. Scope of this conceptual design did not permit detailing the loop design, but a loop

of this basic type will permit satisfaction of all conventional 2200 psi loop requirements, i.e., all utilities, supporting facilities, and floor space are supplied and this basic loop design may be detailed using existing proven components.

The out-of-pile equipment which is visualized is capable of handling one in-pile tube 2-in. inside diameter having coolant requirements up to 150 gpm at 2200 psi and a total heat generation up to 3 Mw and maintaining conventional loop water quality. The loop would have excess cooling capacity to permit operation at low temperature about 350°F, at full power generation. This equipment could also be used to serve three or four manifolded 1 in. i.d. test loops having coolant requirements in the neighborhood of 35 gpm each and total heat generation of up to 750 kw each.

Six cubicle areas are supplied for installation of the loop equipment (see Section 5.21).

5.2 Space Allocations for Experimental Equipment

In view of the difficulties in installing, maintaining and modifying loop systems in confined areas, a special effort has been made to allow sufficient space for experimental equipment from the point of view of total square footage, geometrical configuration (e.g., minimum waste space at the "point" of pie-shaped cubicles), accessibility and proximity of the various operating areas.

5.21 The Primary Cubicles

For each loop, over 750 sq ft of floor space of approximately rectangular geometry with a reasonable ratio of maximum to minimum dimension has been allocated for the loop primary cubicle. Each cubicle area contains a minimum square area of 28 ft x 28 ft. This is over 1-1/2 times the area occupied by the KAPL F-10 loop at the ETR. This area is directly adjacent to the piping corridor. Adequate floor loading will be determined in final design but should be in the order of 5000 lb/ft².

5.22 The Secondary (Sample) Cubicle

At least 300 sq ft of floor space directly adjacent to the primary cubicle has been allocated for a secondary cubicle to be used for the sample and the pressure instrumentation area. These areas have a minimum dimension of 10 ft on their shortest side. It is considered that the wall of the primary cubicle which is common to the secondary cubicle is accessible for primary cubicle reach rods.

5.23 Auxiliary Unshielded Equipment

At least 200 sq ft of floor space per loop has been allocated for make-up and other unshielded equipment. This space is located in the neighborhood of, but not directly adjacent to the cubicle area.

5.24 Control Panels

Space is allocated for at least 30 linear ft of control panel per loop, on the same floor and in the neighborhood of the cubicle area. Suitable service and access aisles have been provided. An additional minimum of 75 sq ft of floor space has been allocated for electrical switchgear.

5.3 Supporting Facilities

Proper supporting facilities and utilities are extremely important to successful operation of a test reactor. This is particularly true during the early months of operation when oversights and inadequacies can become painfully obvious. Every effort has been made to supply all those items essential to the experimental program. (Since the discussion of the data processing system is presented in some detail, it is given in a separate section, Section 5.4.)

5.31 Analytical Laboratory

Sponsor laboratory space must be provided in the ATR building or in adjacent existing buildings.

5.32 Hot Cell

The hot cells already provided for use of MTR and ETR experimental programs will be used for ATR hot cell work.

5.33 Experimenters Canal

A section of canal for the exclusive use of the experimenters has been supplied. This canal is connected with the canal complex and can be isolated from reactor operational activities.

5.34 Mockup and Pressure Test Area

The MTR and ETR have a mockup area in the reactor services wing of the MTR. This will be used for mockup and pressure tests of ATR equipment.

5.35 Cranes

Two fifty-ton cranes with fine control on all three directions of motion are available in the reactor area and in the main canal area for handling of experimental casks. Five-ton cranes or lifts are supplied in the cubicle area. Forty-five feet of headroom is specified in conceptual design for the heavy cranes. It is anticipated that in final design ample headroom will be allowed for discharge procedures planned at that time with a generous contingency for future discharge procedure modifications. A two-ton high crane is also included for the reactor area.

5.36 Warehouse

Warehouse facilities are extremely important since inadequate warehouse space results in inefficient operation of the experiments due to parts shortages. The warehouse under construction in the MTR-ETR area contains conventional warehouse space, an isolated warm storage area for storage of contaminated reusable equipment and a cask storage area with facilities for handling and storing thirty-ton casks.

5.37 Drafting and Reproduction

Necessary drafting and reproduction facilities for sponsor use will be available from existing facilities and organizations.

5.38 Cask Transportation

Facilities are supplied in the conceptual design for loading casks on trucks in the reactor building. Existing facilities will be used for long-haul transportation.

5.39 Hydraulic Test Facility

Hydraulic testing for determining total flow and flow distribution within experimental test assemblies will be done in existing MTR-ETR facilities.

5.310 Darkroom

Darkroom facilities for processing photographs and radiographs are available in the existing MTR-ETR plant.

5.311 Decontamination Room

A decontamination room for decontamination of experimental equipment is provided. This facility has been found very useful at MTR.

5.312 Communications System

The location of all experimental equipment for a given loop in one general area reduces the demands on the communications system but it is important that all areas containing experimental equipment be able to communicate with each other and with the reactor control room. The important outlying areas would include the reactor top, the tank nozzle area, the piping corridor and the sub-pile room.

5.313 Waste Disposal

The methods of waste disposal are a stack, underground storage and a burial ground for gaseous wastes, liquid wastes and contaminated or radioactive hardware, respectively.

Facilities for liquid waste storage and processing are sized on the basis of 250 gal loops. It is visualized that the initial fill of contaminated water in the loop could be cleaned up partially by ion exchange systems. A 1000 gal catch tank is provided for collecting experimental wastes. Depending on activity, the waste can then be routed to the ATR leaching bed or to the MTR-ETR hot waste disposal system. At MTR-ETR, provisions are made to clean up the water with the ETR resin columns, to store it temporarily in underground hot storage tanks, or to ship it to CPP for permanent storage.

5.4 Data Processing System for Experiments and Reactor Operations

The Data Processing System will consist of a high-speed-computer-base data system together with the required remote input stations located at each experiment area. The reactor will have a complete system for logging all required data; one electronic computer will carry out the computations. The accuracy of this system will in all cases be equal to or better than that which is available in analog instrumentation for similar measurements.

The reactor will have about six input stations located near the reactor process equipment and at the experimental facilities to receive up to 300 points of desired information from the reactor and its experimental facilities. The input stations will include precision temperature controlled junction boxes for thermocouple signals, pneumatic to electric transducers for any variable measured with pneumatic type instrumentation, and normal input terminals for electric signals already scaled to the data system input.

The Data System will provide the following functions: off-normal alarm scanning, output data logging on electric typewriters and computing.

5.41 Off-Normal Alarm Scanning

All inputs will be continuously scanned at a rate of 100 points/sec for any off-normal condition as compared to preset values. If an alarm condition is detected, an identification light will come on as well as an audible alarm. When the channel returns to normal, the system will log out on an off-normal printer the time the channel went off-normal, the time it returned to normal and the maximum or minimum value attained during this time.

5.42 Data Logging

The system will log all channels in direct engineering units at various pre-selected time intervals. These recordings will be made on electric typewriters at a rate of 100 channels/sec from the system with adequate buffer capacity so that the typewriters will operate at 2 channels/sec. In addition to the complete normal logging cycle, the system will contain a feature whereby any portion of the system inputs can be logged out on demand at any logging station. There will be logging stations at each major experiment control panel.

5.43 Computations

The system will contain a digital computer whereby normal operational computations such as heat flux, power generation, and fuel burnup can be computed from any input variable as well as manual entry constants. The selection of the manual entry constants in some cases will be a function of the input variables.

5.5 Utilities

The utilities are described in detail in Section 4.8. Only a few remarks need be made here.

5.51 The HDW System

The HDW (High Pressure Demineralized Water) system is designed primarily for the design shown in Fig. 5.1A, i.e., for a water-to-water heat exchange system. It does not preclude use of intermediate heat exchange media, but use of organic intermediates is a more expensive means of heat removal. The heat removal capacity of the HDW system is 16 Mw which closely represents the maximum capabilities of the loops and also allows for future expansion or conversion of capsule facilities to loops.

5.52 Emergency Cooling Loop (ECL) System

The ECL system is designed to have about 1-1/2 times the flow and heat load capacity of the ETR ECL. The reader may easily determine the many combinations of fueled and unfueled in-pile sections which this system can carry.

5.53 Failure-Free Power

One important provision that may have been under-estimated is that of failure-free power. One tends to hold down the capacity of such power which is provided because of the high capital cost per kilowatt. This requirement should be studied very carefully in the final design.

Electrical power loads on critical functions must be split between failure-free and commercial power sources to ensure operation if either power source fails. The two portions of each load must be completely decoupled in an electrical sense to prevent interaction between the two power sources.

5.54 Miscellaneous Utilities

It is not felt that more needs to be said about commercial power, instrument power, instrument air, low-pressure demineralized water and utility cooling water (cooling tower water) than has been said in Section 4.8 except to make the statement which may be obvious; namely, that UCW may not be used as a substitute for HDW in water-to-water systems because of the stress corrosion problem.

5.55 The Cubicle Exhaust and Vent Header Systems

Cubicle exhaust and vent header systems are provided. The vent header system into which the experiment's pressure relief devices discharge is similar to the ETR system. A word about the design criteria for the cubicle exhaust system is in order, however. A total of 57,000 scfm will permit removal of 75 kw of loop heat as well as maintaining the required 150 ft/min air face velocity over openings in the cubicle.

6.0 HAZARDS EVALUATION

6.1 Summary

The reactor proposed in this report utilizes design concepts which differ from those in existing testing reactors. This section considers only hazards that might arise from the unique features incorporated in the ATR. Briefly, the areas where differences in behavior might be expected are:

1. Kinetics. Some differences are suggested because of the core material and its arrangement. The effects of these differences on the kinetics are discussed extensively in Section 6.21.
2. Transient Heat Transfer. Since portions of the core will operate at conditions nearer burnout than in ETR, for example, the consequences of accidents involving loss of primary coolant flow and pressure, power overshoots, and accidental reactivity insertions are evaluated in Section 6.22.
3. Relation of Normal Radioactive Release to Site Location. Because there will be larger normal production of radioactive material (effluent fluids and hot waste) resulting from the operation of this plant than in the MTR and ETR, the site location with respect to disposal facilities is considered in Sections 6.31 and 6.32.
4. Abnormal Release of Fission Products. An accident involving such a release is considered in terms of the same occurrence in ETR in Section 6.33.

A summary of the results of these studies is given here. The kinetic properties of the reactor result from the slowing down and diffusion times of neutrons in the D₂O reflector proposed. The effect of this holdup of neutrons is similar to the effect of the delayed emission of neutrons in the fission process, the differences here being two-fold: 1) the delay time is of the order of 1 msec, and 2) the number of such neutrons per fission is much larger than that of any known delayed neutron emitter.

In their study of single annulus reactors, the Internuclear Company¹ concluded that the effect of dropping the entire D₂O reflector was such as to produce a 30 per cent change in the reactivity. Roughly speaking this means that approximately 30 per cent of the neutrons, which would otherwise have been lost to the system, are returned by the D₂O reflector when it is in place. The kinetic studies which follow have used this figure of 30 per cent total leakage to demonstrate the properties that result.

1. C. F. Leyse et al., "An Advanced Engineering Test Reactor Design", Internuclear Company, Inc., Clayton, Missouri, Report AECU-3775, March 15, 1958.

If one considers the average fission neutron which goes out into the reflector, where it becomes thermalized and drifts back into the core, it is possible to speak of an "arrival" function which represents the average abundance as a function of time of slow neutrons returning to the core per fission event. Although the exact form of the arrival function is difficult to calculate, the kinetic behavior in the region of interest is not extremely sensitive to the particular form chosen. This has been demonstrated in the calculations of the transfer functions for several different arrival functions given in Section 6.21. It was found there that describing the behavior in terms of an equivalent seventh delayed neutron group was sufficiently accurate throughout the entire range of interest. On this basis it was found that the periods which are obtained in this system for a given reactivity step are much longer and hence more easily controlled than those of ETR. Furthermore, because of the existence of negative void coefficients in the light water cooled fuel section, shutdown effects similar to those seen in BORAX will probably provide self-limiting behavior. In this respect ATR appears to have the desirable long lifetime features of heavy water reactors and the prompt shutdown feature of light-water moderated systems.

In the case of loss of flow accidents, the time lapse between loss of flow and burnout is calculated to be more than 2 sec. The instruments which monitor reactor coolant flow initiate a full scram at some pre-set flow rate and if it is set at 90 per cent of full flow the safeties will begin the reactor shutdown in approximately 750 msec after loss of flow. (The scram which arises from loss of voltage to the pumps is actuated in a total of 350 msec.) Thus it seems that there is adequate time for the reactor power to be reduced to a negligible value in case of pump or motor failure. These calculations are based on actual measurements of the "coast down" of the ETR coolant flow.

The calculations involving loss of primary coolant pressure were based on the formation of a 36 sq in. hole in the pressure vessel, somewhere above the core. Under these conditions the time lapse to burnout was calculated to be more than 7 sec. A reactor scram will be initiated by the monitoring instruments in less than 2 sec, which means that for this accident also, no damage to the reactor should occur.

Power overshoots can result from insertions of reactivity introduced either as ramps or steps. Detailed calculations have been performed using the reactor kinetics of the proposed reactor core and its heat transfer characteristics. The maximum step reactivity insertion which will not burn out the core, when operating at full power, is slightly above 0.3 per cent $\Delta k/k$. This maximum permissible step is only slightly less than the equivalent step for ETR, and therefore appears to offer no unusual problem. From the above considerations, it appears that the reactor proposed in this report can be operated without any undue risk and can be protected adequately against the accidents resulting from malfunctions of reactor components or experiments.

The volume of radioactive effluent and hot waste from the reactor will require either its location near an authorized disposal area or the establishment of another such area.

The accidental release of an amount of fission products equal to 1 per cent of the core fission product inventory has been evaluated, and its associated hazard has been found to be similar to that of ETR.

6.2 Reactor Behavior

6.21 Reactor Kinetic Properties

The factor in ATR most likely to influence the short-term time response (and, therefore the high-frequency response also) is the large D_2O reflector. Neutrons which migrate into the reflector and return some time later to cause fissions exhibit properties similar to an extra group of delayed neutrons. The high bare leakage effect makes this a large group (as much as 30 per cent), with a time constant of about 1 msec. For the calculations, a seventh group with a delayed neutron fraction, β_7 , of 0.3 and with an equivalent decay constant, λ_7 , of 1000 sec^{-1} is used. Although methods exist for calculating both the frequency and time responses for six delayed groups, adding the seventh extremely large group produces its own difficulties.

The step-reactivity-time calculation is relatively easy to do, as is also the stable period (in-hour) reactivity relationship. If one assumes that there are six groups with a total delayed neutron fraction of $\beta_T = 0.0075$, plus an extra group of $\beta_7 = 0.3$, calculations reveal somewhat startling results. True prompt criticality now requires a 30 per cent reactivity insertion. Therefore, large (> 1 per cent) Δk excursions do not produce the short periods seen in a light water moderated and reflected system. Because of the short time constant of the seventh group, the behavior for small (< 0.3 per cent) Δk insertions is more like the conventional system except for a very short time after the insertion. After $1/\lambda_7$ seconds have passed, the transient power vs. time is only about a factor of two under the ordinary six delayed group case. The important fact, however, is that the power is truly less than that for the six-group solution at any time, and the stable periods produced by step reactivities for the seven-group case are always longer than those for the six-group case. The results of some of these calculations are displayed as Figures 6.2A, 6.2B, 6.2C, and 6.2D, where Figures 6.2A through 6.2C are comparisons of stable periods for the six- and seven-group cases and in which the prompt neutron lifetime, l , and λ_7 have been varied as parameters. These curves show the ratio of T_7/T_6 , the stable periods of the two cases, versus reactivity necessary for those periods to be produced. Figure 6.2D shows the time-power curves for three cases of core kinetic parameters and two reactivities. These also confirm that the stable and transient (short time) periods are longer for the seven-group cases, and that the resultant power of these reactors subjected to such reactivity steps is always lower than that of the reactor with no delayed seventh neutron group.

It has been the usual procedure in the past to set up a problem for either the ramp (runaway startup accident) or step-transient cases and solve the kinetic equations numerically with a digital computer program. The usual procedure is to allow $k(t)$ to follow a ramp or step up to the time at which a level scram--pre-set at some $n(t)$ --is assumed to insert the safety rods to reduce reactivity at a given rate. This will begin only after some small delay has been overcome; this delay being, for example, the release-time of an electromagnet employed to hold safety rods out of the core, as in the MTR. Estimated release time for the ATR has been used in this report. The numerical integration programs in use, however, do not readily lend themselves to the seven-group cases because of the way the computer program works. When this was discovered, a different attack was used. If the reactor kinetic equation in integral form is expressed:

$$n(t) = 1 + \int_0^t G(t-\tau) K(\tau) n(\tau) d\tau,$$

where G is a function of λ , β , and ℓ , a comparison of two of these integrals can be made:

$$n(t) = 1 + \int_0^t G(t-\tau)_7 K(\tau) n(\tau) d\tau = \\ 1 + \int_0^t G(t-\tau)_6 K(\tau) n(\tau) d\tau,$$

wherein the parts under the integral are made numerically equal by adjusting the lifetime associated with the six group kernel. Solution of two cases (shown in Fig. 6.2E) shows that the seven-group transient case of prompt neutrons lifetime ℓ_7 of 5×10^{-5} sec behaves much the same as a six-group case in which the prompt lifetime is assumed to be 3.3×10^{-4} sec.

Using this method of normalization, the complete time transients were computed for step transients of 0.3, 0.4 and 0.5 per cent reactivity. These curves are drawn in Fig. 6.2F. These curves are the ones from which some of the calculations in Section 6.22 were made. Calculations of the "startup accident" using the same parameters as in those of Fig. 6.2F appear in Fig. 6.2G for three possible ramp rates.

Since the reactor will have an automatic power level controller the frequency response was also calculated and appears in Fig. 6.2H for several cases. The curves of reactor transfer function amplitudes and their phase angles are shown compared with the ordinary six-group case. Cases are calculated assuming the seventh group is simply delayed 10^{-3} sec with no decay (Case 2, Fig. 6.2H), and is delayed 10^{-4} sec and decays as $e^{-\lambda_7 t}$, where $\lambda_7 = 1000 \text{ sec}^{-1}$ (Case 3, Fig. 6.2H). These curves reveal two bits of information. The first is that the amplitude response of the seven-group reactor is lower than that of the six-group case in the region of interest, and secondly, that the most pronounced effect of case 2 is at very high frequencies.

The resonant peaks in one set of the transfer function curves result from Case 2 above wherein no decay is allowed, but the reflector neutrons are simply returned after t sec. It is believed that this does not truly describe the situation and is shown only for academic interest and for comparison with the other cases. Selection of an appropriate model for such calculations is always difficult.

Some conclusions that may be drawn from the investigation, however, are listed below:

1. The reactor is less sensitive to small reactivity changes, i.e., it is "less touchy".
2. A low performance (and therefore less expensive) power level controller can be used successfully on this reactor.
3. The poisoned reflector type of reactivity shimming can be used in the day-to-day operation.

In high-flux thermal reactors the possibility of xenon instability is present under certain conditions. The proposed reactor is one in which the coupling between the various parts of the core is certainly less than it is in the MTR, for example. It is felt, however, that the radial stability can be maintained by the shim control system. The problem of axial stability must be thoroughly investigated when the engineering design is begun.

6.22 Transient Heat Transfer

The reactor concept proposed presents transient problems which have not been duplicated by any operating research reactor. To meet the specifications in the testing facilities, the reactor must operate closer to the burnout point. The purpose of the transient heat transfer calculations is to define the margin from burnout that can be expected and to determine if presently available instrumentation will adequately protect the reactor.

There appears to be two consequences of reaching the burnout condition: 1) release of fission products and fissile material to the primary coolant, and 2) the possibility of an aluminum-water reaction with a resulting large release of energy. The admission of minute quantities of fission products and fissile materials into the primary coolant does not constitute a hazard since the light water cleanup system will adequately purify the water; this comment is based on ETR and MTR experience where such accidents are relatively commonplace. However, the burnout of a portion of a fuel plate probably results in the release of radioactivity several orders of magnitude greater than that involved in the ETR and MTR experiences. It is obviously prudent to avoid the burnout condition during any event. Operation in the nucleate boiling region during a transient does not

present any hazard since the consequences of such an activity do not appear to be hazardous. The thermal shock that the fuel plates undergo during nucleate boiling operations is not severe apparently as evidenced by experience in the ORR and the MTR.

There are several accidents which are considered credible. They are: 1) loss of primary coolant flow, 2) loss of primary coolant pressure, and 3) an accidental power overshoot consisting of either a ramp insertion of reactivity at startup with failure of all but the top level scram or a step insertion of reactivity while operating at full power.

The loss of flow accident generally occurs because of a fault in the electrical power supply to the site. Other possibilities include the overloading of one primary circulating pump and its subsequent stoppage, failure of the main flow control valve and accidental closure or failure of one of the main block valves.

Loss of primary coolant pressure may be caused by rupturing any of the primary components, cracking a weld or the like.

An inadvertent raising of the reactor power by the operator could cause a power overshoot. Increasing the reactor power too rapidly may also result in an overshoot.

The stepwise positive addition of reactivity may occur during the removal or insertion of an experimental assembly while the reactor is operating at full power.

Burnout heat fluxes for all cases were calculated by the extended method of Bernath¹. The set of equations is given below.

$$(Q/A)_{BO} = h_{BO} (T_w - T_b)_{BO} ,$$

where

$$(Q/A)_{BO} = \text{the heat flux at burnout, PCU/hr-ft}^2 ,$$

$$T_{wBO} = \text{wall temperature of the fuel element at burnout, } ^\circ\text{C, and}$$

$$T_{bBO} = \text{bulk water temperature at burnout, } ^\circ\text{C.}$$

h_{BO} , the film coefficient at burnout in PCU/hr-ft²-°C is given by:

$$h_{BO} = 10,980 \left[D_e / (D_e + D_i) \right] = 48v / (D_e)^{0.6} ,$$

where

$$v = \text{coolant velocity, ft/sec,}$$

1. L. Bernath, "A Theory of Local Boiling Burnout and Its Application to Existing Data", proceedings of 3rd National Heat Transfer Conference August 1959.

D_e = hydraulic diameter, ft, and

D_i = heated diameter = heated perimeter/ π , ft.

T_{wBO} is given by:

$$T_{wBO} = 57 \ln P - 54 P / (P + 15) - v/4,$$

where

P = pressure at the burnout point, psia.

Heat fluxes by this method were multiplied by 0.6 to allow for error in fitting to the data. The calculation, therefore, is probably conservative.

6.221 Loss of Flow Accidents

The calculations for this incident involve determining the velocity and corresponding bulk water temperature at which the burnout heat flux is equal to the operating heat flux. The burnout heat flux as a function of velocity is calculated, and from the flow coastdown curve and these calculations the time to burnout after power failure is determined. Many flow coastdown characteristics for the ETR system have been measured using rapid response instrumentation. These curves were normalized and taken as representative for the ATR. This coastdown curve is presented in Fig. 6.2I. It has been found that the fraction of operating flow at times greater than 1 sec after pump failure in ETR is independent of the operating flow. The measurements in the ETR should represent a good approximation of the conditions in the ATR because the primary coolant systems are expected to be similar.

The results of these calculations are summarized in Table 6.2A, and in Figure 6.2J. The time elapsed between loss of flow and burnout is greater than 2 sec. Shutdown of the reactor by means of a full scram can be accomplished within this time increment using commercially available instrumentation. The electrical power frequency can be sensed and a scram initiated following a fault in the electrical supply in 350 msec. Actual measurements of the time required to activate the frequency devices in service at ETR have established this number. Commercially available differential pressure measuring devices are also used to protect the reactor in the event of loss of flow. The differential pressure across the reactor core is used as the primary measurement; a scram is initiated in 1 sec by these devices; this time increment is again based on measured values.

It is concluded that the ATR can be adequately protected from a loss of flow accident. It is possible that some nucleate boiling will occur during the transient, but this is not considered to be hazardous.

TABLE 6.2A

COMPARISON OF OPERATING CONDITIONS AND BURNOUT CONDITIONS

Operating power level, Mw	250
Radial max/avg power	1.87
Channel hydraulic diameter, ft	0.01258
Inlet pressure, psig	300
Coolant velocity, ft/sec	42.5
Nominal hot channel heat flux in millions of Btu/hr-ft ²	1.115
Nominal hot spot heat flux in millions of Btu/hr-ft ²	1.56
Burnout heat flux at normal operating conditions in millions of Btu/hr-ft ²	3.67
Ratio of burnout power at normal operating condition to normal operating power	2.35
Hot spot heat flux at burnout power level (caused by power excursion, including effect on bulk temperature) in millions of Btu/hr-ft ²	2.37
Ratio of burnout power (caused by power excursion) to normal operation power	1.52
Time to burnout after pump failure, sec	2.6
Coolant velocity at burnout resulting from pump failure, ft/sec	26

6.222 Loss of Pressure Accident

In order to evaluate this accident the time delay from the rupturing of the pressure vessel, above or below the core, or the inlet or outlet coolant lines to burnout is calculated from the data of Table 6.2B and the assumptions of Table 6.2C. As shown below, the ATR is adequately protected from a loss of pressure accident.

Pressure at Which Burnout Occurs. The Bernath burnout correlation, minus a 40% safety factor, is used to calculate the pressure at which burnout occurs.

$$\begin{aligned} (Q/A)_{BO} &= (Q/A)_{\max} = 1.56 \times 10^6 \text{ Btu/hr-ft}^2 \\ &= (0.6) (1.8 \text{ Btu/PCU}) h_{BO} (T_{w_{BO}} - T_{b_{\max}}) \end{aligned} \quad (1)$$

$$T_{w_{BO}} = 57 \ln P_{BO} - 54 P_{BO}/(P_{BO} + 15) - V/4 \quad (2)$$

T is in °C
P is in psia
V is in ft/sec
h is in PCU/hr-ft²-°C

Substitution of Equation 2 into Equation 1 yields:

$$(Q/A)_{\max} = (0.6)(1.8)(h_{BO}) \left[57 \ln P_{BO} - 54 P_{BO}/(P_{BO} + 15) - V/4 - T_{b_{\max}} \right]. \quad (3)$$

Since all other quantities in Equation 3 are known (see Table 6.2B), P_{BO} may be calculated. The pressure existing above the core when burnout occurs is $P_{BO} + \Delta P_{\text{hot spot}}$. As shown in Fig. 6.2K, burnout occurs at 117 psig.

System Pressure Change After Rupture, as a Function of Time. The water lost from the system as a function of time, which is also the increase in air volume in the surge tank is given by:

$$V_{t+\delta t} = \delta t A \sqrt{2\Delta P g} + V_t, \quad (4)$$

where δt is a time interval in sec.

The initial pressure above the core is essentially the same as the pressure in the surge tank, i.e., 312.5 psia or 300 psig. The pressure-volume relationship of the air is given by the perfect gas law:

$$PV = nRT. \quad (5)$$

Since n , R and T are all constant the pressure, at any time, is related to the initial pressure by:

$$P_t = P_o \left(\frac{V_o}{V_t + V_o} \right), \quad (6)$$

TABLE 6.2B
SYSTEM DATA AT NORMAL OPERATIONS

P_{inlet}	300 psig = 312.5 psia
$T_{w_{max}}$	424°F (calculated from hot-spot hot-channel analysis)
$P_{hot\ spot}$	250.5 psia
$(Q/A)_{max}$	1.56×10^6 Btu/hr-ft ²
D_e	0.01258 ft
D_i	0.197 ft = heated perimeter/ π
v	42.5 ft/sec in the core
h_{BO}	$10,980 \frac{D_e}{D_e + D_i} + \frac{48v}{D_e^{0.6}} = 28,800$ PCU/hr-ft ² -°C (Bernath correlation)
$T_{b_{max\ @\ hot\ spot}}$	135°C (calculated from hot-spot hot-channel analysis) = 275°F
$T_{b_{inlet}}$	130°F
$\Delta P_{hot\ spot}$	312.5 - 250.5 = 62 psi
Volume of surge tank	4000 gal
Volume of air in surge tank	2667 gal = 356.7 ft ³

TABLE 6.2C

ASSUMPTIONS USED IN CALCULATING LOSS OF PRESSURE ACCIDENT

1. The maximum credible loss of pressure accident would be a rupture in the inlet line or the pressure vessel above the core.
2. A complete fracture and displacement of the inlet line is not considered credible.
3. A fracture resulting in a 36 in.² hole in either the pressure vessel above the core or the inlet coolant line will be considered as the maximum credible accident.
4. The loss of pressure rate will be the same for the same size fracture in the pressure vessel above the core as for the inlet coolant line fracture.
5. The loss of coolant through the fracture may be calculated by assuming that the hole is equivalent to an orifice of 36 in.² area and using the following equation:

$$V = AC \sqrt{\Delta P(2g)}$$

V = volume rate of coolant, ft³/sec

C = discharge coefficient = 1

ΔP = pressure differential across the fracture in ft of H₂O

g = gravitational constant

A = area of fracture, ft²

6. The coolant velocity through the core, the core power and the inlet coolant temperature remain constant during the loss of pressure; in fact, the coolant velocity decreases only slightly.
7. The surge tank responds instantaneously to any change in pressure of the coolant system.
8. The water and air in the surge tank are at the same temperature and pressure as the coolant entering the reactor.
9. The perfect gas law is correct for calculating P vs. V for air.
10. The effect of the system pressurizer pumps is neglected. (It takes 3 sec for the pressure loss to be detected and for the back pressure valve to close and the pumps to start.)

where V_t is the volume of water lost up to time t as determined by Equation 4 and $V_t + V_o$ is the volume of air at time t .

The volume of coolant discharged is calculated from Equation 4 starting at the time of rupture and using small time increments. The resulting pressure is calculated from Equation 6. Burnout occurs when P_t equals the pressure calculated to exist above the core at burnout ($P_{BO} + \Delta P_{hot\ spot}$). Pressure is plotted as a function of time in Fig. 6.2K and the time delay from rupture to burnout may be read directly from the curve. The figure indicates a 12 sec delay at full reactor power. A reactor scram can be initiated in about 2 sec, because the pressure falls about 11 per cent in the first second and thus actuates the pressure sensing instrumentation which has a time delay that should be somewhat less than 1 sec. Therefore, the reactor is adequately protected from a loss of pressure accident.

6.223 Accidental Power Overshoot

The accepted practice is to set the neutron safety levels at 1.3 times the full power level (N_f). This power level is below the burnout value at operating coolant velocity and inlet temperature. The burnout flux is compared with the operating flux in Table 6.2A as an indication of the margin of safety during operation. Two possible methods of causing a power overshoot have been analyzed, the power overshoot resulting from a ramp insertion of reactivity at startup and that resulting from a stepwise insertion of reactivity while operating at full power. These situations represent the most severe accident believed credible. The nature of these power transients is discussed in Section 6.21.

Ramp Insertion of Reactivity. The ramp accident assumes that at source power (criticality) the rods begin driving in as a gang and that all scrams except the highest level scram, i.e., 1.3 N_f , fail. The relative power as a function of time is given in Fig. 6.2G for ramp insertions of 0.10, 0.15 and 0.20 per cent $\Delta k/k/sec$. The scram level, shown as a relative power of 10^6 , is 1.3 N_f or 2.03×10^6 Btu/hr-ft² at the hot spot. Burnout occurs at 2.37×10^6 Btu/hr-ft² or at a relative power of $(2.37 \times 10^6 / 2.03 \times 10^6) 10^6 = 1.17 \times 10^6$. Figure 6.2G shows that even the 0.10 per cent $\Delta k/k/sec$ exceeds burnout, i.e., $n/n_o = 1.17 \times 10^6$, is less than 0.1 per cent $\Delta k/k/sec$. This value, however, has a safety factor in that the reactor self regulation, i.e., the temperature coefficient, has been intentionally neglected. If the rod drive rates are such that a ramp insertion of 0.1 per cent is possible, lower scrams must be operative to prevent reactor failure.

Stepwise Insertion of Reactivity. The most severe result of stepwise insertion of reactivity occurs at full power. A reactor scram is initiated at the upper level scram setting of 1.3 N_f .

Power transients produced by 0.3, 0.4 and 0.5 per cent $\Delta k/k$ step insertions are presented in Fig. 6.2F. The effect of 0.3 and 0.4 per cent step insertions on the reactor cooling has been calculated assuming that: 1) the burnout condition predicted by the Bernath

equation represents the condition in which the boiling film completely blankets the surface, and 2) after burnout occurs negligible heat is transferred. The hot spot heat flux, at any time during the transient, is obtained from the normal operating hot spot heat flux by multiplying by the power ratio existing at the time in question.

$$Q/A_{\text{hot spot}}(\theta) = n/n_o(\theta) Q/A_{\text{normal hot spot}} \quad (7)$$

The hot channel bulk water temperature rise is obtained as follows:

$$\Delta T_{b \text{ transient}} = \Delta T_{b \text{ operating}} \left[\frac{\int_{\theta - \theta_t}^{\theta} \frac{P\theta}{P_o} d\theta}{t} \right] \quad (8)$$

where

θ = time,

P/P_o = ratio of power at time θ to the normal operating power (Fig. 6.2F),

θ_t = transient time of a water particle from inlet to hot spot, and

ΔT_b = hot channel bulk water temperature rise from inlet to hot spot.

Using Bernath's equation, with the bulk water temperature calculated for the hot spot as shown above, the burnout heat flux may be calculated at any time during the transient. Burnout occurs when the hot spot heat flux equals the burnout heat flux. After burnout occurs and until the hot spot heat flux is again less than the burnout heat flux, all of the heat is assumed to be absorbed by the plate thereby increasing its temperature. The peak temperature of the plate is given by:

$$T_{\text{peak}} = T_{w_{BO}} + \int_{\theta_{BO}}^{\theta} \frac{Q/A(\theta) d\theta}{C_A} \quad (9)$$

where

C_A = the heat capacity of a square foot of plate surface,

Q/A and $(Q/A)_{BO}$ are determined from Equations 3, 7, and 8 and are shown in Fig. 6.2L and 6.2M for 0.3 per cent and 0.4 per cent step insertions.

In the transient produced by the insertion of 0.4 per cent reactivity, the burnout condition is reached in about 80 msec and Equation 9 indicates a peak temperature of 1400°F; this temperature is above the melting point of aluminum. Therefore,

a step insertion of reactivity of 0.4 per cent during operation at full power cannot be tolerated. Burnout is not reached when 0.3 per cent $\Delta k/k$ is inserted. The insertion of 0.3 per cent positive reactivity would not result in either a release of fission products into the primary coolant or a metal-water reaction and should therefore be safe. This value has the same safety factor as the ramp insertion, i.e., the reactor self regulation due to its negative temperature coefficient has been purposely neglected.

6.3 Effluent Control and Area Location

The Phillips Petroleum Company Report IDO-16623 by D. R. deBoisblanc, R. J. Nertney and L. H. Jones shows that the location of ATR adjacent to MTR-ETR meets all safeguards requirements. Also, as discussed in Section 4.8 of this report, adequate facilities will be provided for discharge of liquid, solid and gaseous wastes in accordance with NRTS practice.

FIG. 6.2 A
 RATIO OF STABLE PERIODS vs. REACTIVITY STEP

FIG. 6.2 B
 RATIO OF STABLE PERIODS vs. REACTIVITY STEP

FIG. 6.2 C
 RATIO OF STABLE PERIODS vs. REACTIVITY STEP

FIG. 6.2 D
 COMPARISON OF STEP TRANSIENTS FOR TWO REACTIVITIES

FIG. 6.2 E
COMPARISON OF STEP TRANSIENTS FOR SIX AND SEVEN GROUPS

FIG. 6.2 F
STEP REACTIVITY TRANSIENT POWER OVERSHOOTS

FIG. 6.2 G

STARTUP ACCIDENT POWER CURVES FOR THREE RAMP RATES

FIG. 6.2 H

ZERO POWER REACTOR FREQUENCY RESPONSES

FIG. 6.2 I
FLOW DECAY IN ETR

FIG. 6.2 J
BURNOUT IN LOSS OF FLOW ACCIDENT

FIG. 6.2 K

LOSS OF PRESSURE ACCIDENT-PRESSURE vs. TIME FOR 36 in²
BREAK IN THE PRESSURE VESSEL ABOVE THE CORE

FIG. 6.2 L

HEAT FLUX IN CORE DURING POWER TRANSIENT FROM
STEP INSERTION OF 0.3% $\Delta k/k$

FIG. 6.2 M

BURNOUT IN CORE DURING POWER TRANSIENT FROM
STEP INSERTION OF 0.4% $\Delta k/k$

7.0 ACKNOWLEDGMENTS

To G. M. Kavanagh, I. F. Zartman and Doyle Rauch of AEC Nuclear Technology for their guidance and suggestions concerning the scope of this study.

To D. G. Boyer of AEC-IDO for his valuable services in obtaining information and clarifications of project objectives.

Special acknowledgment for their contributions is accorded to the two consultants from the Internuclear Company who worked with the Phillips staff during the project: B. H. Leonard, Jr. (Reactor Design) and T. L. Francis (Reactor Physics).

8.0 APPENDICES

8.1. Quality Control of Boron in Recent ETR Elements

February 26, 1960

Boron Content of ETR Fuel Assemblies
and ETR Control Rod Fuel Sections
Contract C-217
WMH-26-60A

Mr. F. E. Watson, Jr.
General Electric Company
Atomic Power Equipment Department
Post Office Box 234
San Jose 12, California

Dear Sir:

The results of the ETR Critical Facility's measurements of boron in ETR core fuel assemblies and ETR control rod fuel sections recently shipped to us are as follows:

400 g elements

<u>Fuel Element No.</u>	<u>Boron Content (grams)</u>	<u>Fuel Element No.</u>	<u>Boron Content (grams)</u>
U-756	2.12	U-796	2.04
760	2.06	800	2.05
764	2.19	804	2.04
768	2.10	808	2.06
772	2.15	812	2.06
776	2.19	816	2.01
780	2.02	820	2.08
784	2.09	824	2.04
788	2.09	828	2.01
792	2.10	831	3.11

160 g Control Rod Section

<u>Section No.</u>	<u>Boron Content (grams)</u>	<u>Section No.</u>	<u>Boron Content (grams)</u>
U-125	0.78	U-141	0.84
129	0.82	145	0.80
133	0.82	149	0.80
137	0.85		

Very truly yours,

WMHawkins:hj/HK

Manager, Division Engineering
Atomic Energy Division

cc: Messrs. F. E. Watson, Jr.
R. L. Doan, Phillips

8.2 History and Need of Very High Flux Test Reactors

The need for higher neutron fluxes than obtainable in the MTR and ETR has been recognized by the Commission and its contractors for a number of years. Based on experimenters' requests and operating experience, Phillips Petroleum Company replied to a request from the AEC on the needs for advanced testing reactors and on the testing conditions they should fulfill. A report¹ issued in May, 1956, suggested in essence that an advanced test reactor be designed and constructed which provides thermal and fast neutron fluxes in the 10^{15} to 10^{16} n/cm²-sec range, and emphasis be placed on smaller test holes of about 3 in. x 3 in. cross section. More recently J. A. Lane, ORNL,² reviewed the need for very high fluxes and the type of reactors being planned in the United States to obtain these fluxes. He also concluded that neutron fluxes in the 10^{15} to 10^{16} range were needed for isotope production, chemistry studies, cross section measurement and fuel element testing. In the latter part of 1958 the AEC issued a "Request for Proposals to Supply Irradiation Services" to commercial suppliers. Appendix A of this request contained NR test requirements for very high neutron fluxes. These requirements have since been revised and new requirements are reported in an AEC letter from NR to DRD³.

Definitive work on very high flux testing reactors for fuel element experiments was initiated in 1956 when the AEC requested four companies to study, independently, reactor concepts which would provide unperturbed thermal fluxes above 10^{15} n/cm²-sec in certain specific loop type experiments. The specifications called for:

1. an unperturbed flux of 10^{15} n/cm²-sec in a 3 in. and a 4 in., 2000 psia, 500 to 600°F, circulating light water loops,
2. an unperturbed flux of 1.5×10^{15} n/cm²-sec in three 6 in. x 6 in. and one 4 in. x 4 in. versatile gas or liquid metal loops; the gas loop to operate at 2200°F and 300 psia, and the liquid metal loop at 1500°F and 300 psia, and
3. an unperturbed flux of 10^{15} n/cm²-sec in a 3 in. liquid metal loop.

ASTRA⁴ recommended a graphite-moderated and -reflected homogeneous-fuel-type reactor. The large graphite core was penetrated by parallel Zircalloy tubes through which flowed a heavy-water solution of uranyl

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1. J. R. Huffman, W. P. Connor, G. H. Hanson, "Advanced Testing Reactors", Phillips Petroleum Company, Idaho Falls, Idaho, Report IDO-16353, May 28, 1956.
 2. J. A. Lane, "The Design and Need for Ultra High Flux Reactors", ORNL, presented at Symposium on Experience in the Use of Research Reactors, Harwell, England, June 11, 1959.
 3. I. H. Mandil, "Revised NR Requirements for an Advanced Test Reactor", Letter to G. M. Kavanagh, AEC-DRD, August 28, 1959.
 4. R. G. Mallon, J. Saldick, R. E. Gibbons, "Conceptual Design of an Advanced Engineering Test Reactor", Advanced Scientific Techniques Research Associates, Milford, Connecticut, Report NYO-4849, March 1, 1957.

sulfate. Total reactor power was estimated to be 220 Mw. Nine vertical through-holes of various sizes, having thermal neutron fluxes greater than 1×10^{15} n/cm²-sec were provided. Plant capital cost was estimated at \$30,000,000, and a large development program was inherent in the use of a circulating homogeneous fuel solution.

Aeronutronic Systems, Inc.¹ recommended a homogeneous reactor system. The uranyl-sulfate solution in D₂O was circulated through the core and external circuit. The seven experimental loops were inserted into the center of the spherical reactor. At full reactor power of 500 Mw, the unperturbed thermal neutron flux in the central core region was estimated to be 6×10^{15} n/cm²-sec. Plant capital cost was estimated to be over \$60,000,000, and a large development program was inherent in this design also.

Curtiss-Wright Corporation² recommended a large heterogeneous fuel-type reactor with a heavy-water moderator-reflector. The seven experimental loops were located at the center of the cylindrical core. The reactor power varied from 520 to 360 Mw over an operating cycle to produce a usable thermal neutron flux ranging from 1×10^{15} to above 2×10^{15} n/cm²-sec.

Internuclear Company^{3,4} recommended a complex of seven small separate "flux-trap" type reactors contained in a single large biological shield. Each reactor consisted of a cylindrical annular core of aluminum plate-type fuel elements (MTR-type), cooled with H₂O, reflector-moderated outside the core with D₂O, with H₂O annulus just inside the core and a central test loop. The inner H₂O annulus thermalizes the fast flux causing the thermal flux to peak in the water annulus and providing a high thermal flux in the test region.

A single size reactor was recommended for all loops. At a total power of 500 to 900 Mw, the unperturbed thermal neutron fluxes of 1 to 1.5×10^{15} n/cm²-sec were obtained in the test loops. Reflector control was suggested whereby the outer portion of the moderator reflector was poisoned with boric acid for shim control and a pure inner portion could be dropped for safety control.

In a continuing effort on the flux-trap type reactor, Internuclear Company⁵ in 1957-1958 prepared a conceptual design to fulfill the test

1. "A Selection Study for an Advanced Engineering Test Reactor", Aeronutronic Systems, Inc. (Subsidiary of Ford Motor Company), Glendale, California, Report AECU-3478, March 29, 1957.
2. "A Conceptual Design Study of an Advanced Engineering Test Reactor", Curtiss-Wright Corporation, Research Division, Quehanna, Pennsylvania, Report CWR-464 (Del.) May 1, 1957.
3. O. J. Elgert, C. F. Leyse, D. G. Ott, "Preliminary Investigation for an Advanced Engineering Test Reactor", Internuclear Company, Clayton, Missouri, Report AECU-3427, February 22, 1957.
4. C. F. Leyse, B. H. Leonard, *ibid*, Report AECU-3427 (Add.), April 16, 1957.
5. C. F. Leyse, et al., "An Advanced Engineering Test Reactor Design", Internuclear Company, Clayton, Missouri, AECU-3775, March 15, 1958.

requirements of the seven loops described previously. At the same time American Standard¹ reviewed the test reactor complex design recommended by Internuclear Company in the original study.

The Internuclear Company conceptual design again consisted of seven, single-annulus fuel, flux-trap reactors. Two core sizes were recommended, one to provide a 7 in. diameter test hole and the other a 4 in. diameter. The total design power for the complex of reactors was calculated to be 980 Mw. Major features of this design were the use of involute-curved fuel plates to provide a uniform core region, thereby reducing hydraulic problems and increasing heat transfer area, and a non-uniform fuel loading in the plates to flatten radial power distribution. Internuclear Company at present is continuing work on this test reactor complex concept under an existing AEC contract.

The National Laboratories have been interested in very high flux reactors for a number of years. Work on the Mighty Mouse Reactor² was initiated by ANL in about 1956. A number of other studies on very high flux reactors have since been reported by the National Laboratories. J. A. Lane's talk, reference 2 on first page of this section, summarizes the high flux test reactors and includes a list of the very high flux reactors being planned in the United States. Those scheduled for construction by the National Laboratories and their completion dates are:

1. B²R², 1962 (BNL),
2. HFIR, 1963 (ORNL), and
3. AHFR, 1964 (ANL).

The B²R² reactor is proposed for beam type experiments. The reactor core consists of ETR type fuel elements, 17 in. long, and is cooled, moderated and reflected with D₂O. The maximum thermal flux may reach 1×10^{15} n/cm²-sec some distance in the reflector.

The HFIR uses the internal flux trap principle to obtain unperturbed neutron fluxes up to 5×10^{15} n/cm²-sec for the production of heavy elements and other isotopes. Design features have not yet been established except that it will be a single-annular-type core about one foot long with a center flux-trap island (similar to the Internuclear test reactor complex described previously).

The AHFR will be used for physical science research. The core employs conventional ETR type fuel elements, has a center flux-trap island, is H₂O cooled and moderated, and is beryllium reflected. A

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1. M. McVey, et al., "Evaluation of an Advanced Engineering Test Reactor Design", American-Standard, Mountain View, California, Report ASAE-S-11, July 15, 1958.
 2. L. E. Link, et al., "The Mighty Mouse Research Reactor Preliminary Design Study", Argonne National Laboratory, Lemont, Illinois, Report ANL-5688, March, 1957.

number of beam tubes terminate in the reflector, a 1.5 in. diameter central thimble is located in the flux-trap island and a number of other smaller experimental facilities are provided. The maximum unperturbed thermal neutron flux is estimated to be about $3 \text{ to } 4 \times 10^{15}$ n/cm²-sec.

Other groups also have studied very high flux reactors. In 1956 a group of students at Oak Ridge School of Reactor Technology prepared a design of a heavy water, heterogeneous-fuel, cylindrical flux-trap reactor¹. Their calculations indicated that a reactor with an 80 cm diameter central hole and operating at 425 Mw could deliver the following average thermal fluxes in 10 cm diameter experiments: in thorium, 5.8×10^{14} ; MTR fuel element, 2.2×10^{15} ; and in bismuth, 3.9×10^{15} . The following year another group of students prepared a design of a double-slab-type flux-trap reactor² which has interesting possibilities, particularly for long cores to accommodate long experiments. The same year, Phillips Petroleum Company made an exploratory study of coolant-moderator and flux-trap materials for very high flux reactors³, which showed their effect on attainable neutron fluxes (and the required reactor operating powers) as a function of the diameter of the central experimental hole and its contained experiment. A symposium on "Ultra High Flux Research Reactors" was held in December of 1957 and a transcript of the ORNL seminars was issued the following year.⁴ In May, 1959, a Test Reactors Meeting for Industry was held in Idaho Falls, Idaho; these papers are recorded in report IDO-16520.⁵

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1. R. D. Cheverton, et al., "High Flux Research Reactor", Oak Ridge School of Reactor Technology, Oak Ridge, Tennessee, Report CF-56-8-206, August, 1956.
 2. W. L. Carter, et al., "Design Study of an Advanced Test Reactor -- Reactor Design and Feasibility Study", Oak Ridge School of Reactor Technology, Oak Ridge, Tennessee, Report ORNL-CF-57-8-5, August, 1957.
 3. R. J. Howerton, G. H. Hanson, W. P. Connor, "Parameters of High Flux Testing Reactors", Phillips Petroleum Company, Idaho Falls, Idaho, IDO-16406, August 23, 1957.
 4. J. A. Lane, Summary of Seminars on "Ultra High Flux Research Reactors", ORNL-CF-58-7-117, July 31, 1958.
 5. "Test Reactors Meeting for Industry", Report IDO-16520, May 13-15, 1959.

8.3 Boric Acid Solution Control - Current Experimental Knowledge

8.31 Introduction

Chemical solution control with boric acid is proposed for ATR, and it will be the first high power reactor employing this method. The selection is based on (1) the successful experience of ANL in operating Borax III and IV for many hours with significant concentrations of boric acid at megawatt power levels and at temperatures and pressures up to Borax and also ATR design values (i.e., up to 420°F and 300 psig) and (2) the extensive body of experimental data developed by Westinghouse and ANL which is the basis for the use of a combination control system for the Yankee Power Reactor involving control rods and boric acid solution and which is also the basis for employing boron solutions for emergency shutdown of naval reactors. It is emphasized that the ATR application of a chemical control solution peripheral to the core and not in the coolant-moderator is a simpler and less exacting application. Although the above-mentioned successful Borax operations and the Westinghouse and ANL research data are very encouraging, additional R and D work is needed to provide specific information for the ATR control system design. Although Westinghouse has not proposed full chemical control for the Yankee Reactor, their experimental data do not rule out (in fact, they support) the chemical control design proposed for ATR.

It is the purpose of Appendix 8.3 to present the conditions under which the ATR control solution operates and to summarize the ANL Borax operations and the Westinghouse research which is applicable to ATR design.

8.32 Operating Conditions for ATR Control Solution

The operating temperature and pressure of the ATR control solution are about 150°F and 300 psig. The maximum energy release in the solution due to the n, α reaction with boron-10 is approximately 40 watts/gram, and the maximum energy release resulting from gamma-ray stopping and neutron slowing down is about 20 watts/gram. At full power operation solutions containing concentrations as high as 50 grams of boric acid per liter may be used. If lower chemical concentration (but the same B^{10} atom density) is indicated, B^{10} enriched solutions may be used. The required flow rate of the boric acid solution for control of ATR is about one foot per second. For containment of the boric acid solution in the reactor aluminum is a promising material.

8.33 ANL Borax Operations with Boric Acid

8.331 Operating Experiences

Boric acid was used in Borax III because there was not enough negative reactivity in its control rods to poison-down the reactor at temperatures less than design during most of the life of the core. With Borax IV its control rods were adequate for studies with the

59 fuel element core. However, during operations with the 69 element core boric acid was also needed to hold down the reactor at temperatures below design. With Borax IV boric acid solutions were present in the reactor at temperatures of 90-100°F essentially continuously during the 6-month period of this investigation; with Borax III the same situation obtained during about the first 6 months of this 10-month study.

With both Borax III and IV, boric acid solutions containing as much as 1.5 grams/gallon were employed during reactor operations at megawatt power levels. The major experience with boric acid solution in an operating reactor was obtained during the heating up of coolant-moderator to the design temperature and pressure of 420°F and 300 psig for each of the major tests in the Borax III and IV series. The reactor was operated at about 1 megawatt during the heating-up period, and approximately four hours were required for this operation. When the reactor reached the design temperature and pressure, the boron was removed from the system by circulation through a strong anion resin bed. This operation also required about four hours. Due to the high heat loss from the poorly insulated pressure vessel, the reactor power during the boron-removal operation was also approximately 1 megawatt. The above event was carried out 10 to 20 times in Borax III and 4 times in Borax IV. These successful experiences constitute the most important support for the use of chemical solution control for ATR.

Boric acid solution was introduced for the reactor shut-down. Although the reactor power was negligible during the cooling down process, operation for quite a few hours at elevated temperatures in a high fission product radiation field represents additional important experience.

More than a score of criticality measurements were made with boric acid in the coolant-moderator, particularly with Borax IV when the 69 fuel element core was built up from the 59 element unit. In connection with each criticality study, the reactor power was at a significant level for about 15 minutes, the maximum power being 200 kilowatts.

8.332 Corrosion

No abnormal corrosion of materials was observed. With Borax III the fuel assemblies were 2S aluminum; with Borax IV the aluminum-nickel alloy X8001 was employed. The same pressure vessel and internal structural components were employed for both Borax cores. The pressure vessel was standard 304 stainless steel. The grid support was aluminum, probably 2S. Therefore, operations at these Borax conditions can use standard 304 stainless and 2S & X8001 aluminum.

NOTE: The ALPR (now designated SL-1) reactor has also logged several hours exposure at megawatt power levels of X8001 aluminum-alloy fuel elements to boric acid solutions.

8.333 pH Control

Deionized water of megohm quality was employed. The pH of fresh boric acid solutions with concentrations of 1 to 1.5 grams of boric acid per gallon was about 5.5. During operations, as a slight concentration of cations (e.g., aluminum and iron) built up in the solution, the pH increased to 6.0 and sometimes to 6.5. The pH was maintained below 6.5, and even below 6.0, by circulation of the solution through a cation resin column; this operation did not remove any boric acid. With aluminum components in the reactor system, the pH range of 5.5 to 6.5 was desirable.

8.334 Stability of Boric Acid Solutions

The stability of the boric acid solutions during the borax investigations was excellent. The boric acid always stayed in solution under this difficult dynamic situation in which water was boiled off and make-up water added continuously. There was no plating-out of boron under radiation and thermal exposure; within experimental tolerances, reproducibility of control rod positions with comparable operating conditions was obtained throughout each study. This is a very sensitive method for detection of residual boron.

8.34 Mechanical and Solution Control of Yankee Power Reactor¹

Control of the Yankee core will be provided by a combination of control rods and a chemical neutron absorber dissolved in the coolant moderator. Twenty-four movable control rods are used to control reactivity at operating temperatures. Space is provided for eight additional fixed shim rods near the periphery of the core, which are to be used if necessary to adjust initial reactivity of the core. Boric acid is added to the coolant-moderator for intermediate temperature range shutdown and to hold the reactor sub-critical in a clean condition at room temperature. The control rods themselves will be programmed to attain improved flux patterns during operation.

The chemical control system is designed to shut down the cold clean reactor by a margin of at least 5 per cent Δk with all the control rods inserted. Calculations indicate that this will require a boron concentration of approximately 950 ppm. The actual boron concentration required will be evaluated during the full core nuclear experiments.

The chemical compound which will be used in the chemical control system is boric acid. Boron compounds have good thermal stability and have adequate solubility in the cold reactor. The solubility of boric acid at room temperature is 50,000 ppm or 8,000 ppm of natural boron. The solubility is, therefore, several times greater than required. A concentration of 1800 ppm is sufficient to reduce k_{eff} to unity at any temperature above 68°F, even though all control rods are withdrawn.

1. Yankee Atomic Electric Co., Westinghouse Electric Corporation, and Stone & Webster Corporation, "Yankee Nuclear Power Station Technical Information and Final Hazards Summary Report," YAEC-167 (Vol. 1 & 2) 1959.

The design makes use of a feed and bleed system to change the concentration of the chemical neutron absorber. The maximum rate of water injection for this system is 100 gpm. Since the control rods can handle xenon and samarium transients, there is no need for faster action by the chemical control system. The following tabulation presents three important values of the effective multiplication factor, k_{eff} , for the Yankee reactor.

<u>Condition</u>	<u>k_{eff}</u>
Cold, clean	1.198
Cold, clean, all rods in	1.044
Cold, clean, all rods in, 950 ppm boron	0.950

8.35 Metallurgical and Chemical Laboratory Research

Argonne did much of the early research which was directed to the use of chemical solution control. These studies are summarized in the classified report ANL-5244¹. Most of the recent investigations in this field have been carried out by the Westinghouse Corporation. The physical properties of boric acid solutions and a chemical evaluation of its use as a shutdown poison are given in Westinghouse-Bettis internal documents WAPD-C(PC)-31² and WAPD-M(CDA)-258³, respectively. The Architect-Engineer will find Reference 2, "A Study of Chemical Control for PWR", a particularly helpful general report to assist in the detailed engineering design of the proposed chemical control system for the ATR.

8.351 Corrosion

The successful Borax operations with aluminum fuel elements and structural components in contact with boiling boric acid solutions make aluminum a promising material of construction for containment of the ATR control solution. The Westinghouse-Bettis Lab has made extensive tests of the corrosion by boric acid of materials used in naval power reactor systems; however, since aluminum is not employed in these reactors, the corrosion of aluminum was not investigated. The Westinghouse Lab supporting the Yankee Power Reactor has studied the corrosion of 304 stainless steel and 4135 carbon steel. These

1. C. R. Breden, U. S. Brown, and M. Sivitz, "Soluble Poisons in Reactor Control", ANL-5244, November 1955 (Classified).
2. "A Study of Chemical Control for PWR", WAPD-C(PC)-31 dated April 8, 1955.
3. J. R. Coulter and K. H. Vogel, "Chemical Evaluation of Potassium Tetraborate and Boric Acid as Chemical Shutdown Poisons", WAPD-M(CDA)-258, April 1960.

Westinghouse data are summarized in the Bettis reports WAPD-M(MEE)-975¹ and WAPD-CP-545². Uhlig's Corrosion Handbook³ and the Shell Company Corrosion Data Survey⁴ give some general information about the corrosion of aluminum by boric acid; but since this information is not referenced, its present utility is limited. Therefore, in order to make final selection of the material and to develop an engineering design for the containment vessel, and also for the final selection of materials for pumps, instruments, etc., the ATR R and D program must provide the specific data needed. The above-mentioned Westinghouse boric acid corrosion investigations, although not concerned with aluminum, provide valuable information about experimental techniques and general boric acid corrosion experience.

8.352 Irradiation Effects

The investigation of fast electron and gamma irradiation of boric acid solutions to dose rates of 2×10^6 rads is presented in Bettis reports WAPD-M(CDD)-149 and 150^{5,6}. The solutions were found to be relatively stable to electron and gamma radiation in both the Van de Graaff and MTR Gamma Facility tests even in the absence of added hydrogen. These data show only H_2 produced by the irradiation. However, during the reactor irradiation of boric acid solutions in ANL CP-3⁷ which represents much more closely the ATR situation, the production of H_2O_2 was almost equal to the H_2 produced. The difference in products is apparently caused by

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1. R. E. Moore and E. Rau, "Corrosion Evaluation of Boric Acid and Potassium Tetraborate as Chemical Shutdown Poisons for Naval Reactor Plants", WAPD-M(MEE)-975, December 2, 1959.
 2. H. K. Lembersky, B. G. Schultz and P. Cohen, "Corrosion of Materials in Boric Acid Solutions Under Dynamic Conditions", WAPD-CP-545, February 1955.
 3. H. H. Uhlig, "The Corrosion Handbook", John Wiley and Sons, Inc., New York, 1948.
 4. G. A. Nelson (Compiler), "Corrosion Data Survey", Shell Development Company, Emeryville, California, 1960.
 5. J. M. Wright and T. R. Druga, "Fast Electron Irradiation of Potassium Tetraborate, Ammonium Pentaborate and Boric Acid Solutions", WAPD-M(CDD)-149, May 5, 1960.
 6. Y. Solomon and J. M. Wright, "Fast Electron and Gamma Irradiation of Potassium Tetraborate, Ammonium Pentaborate and Boric Acid Solutions", WAPD-M(CDD)-150, May 20, 1960.
 7. E. Hart, W. McDonnell and S. Gordon, Proc. Int. Conf. Peaceful Uses of Atomic Energy, 7, 593 (1955).

the difference in irradiating particles. With neutron irradiation which results in the B^{10} (n, α) Li reaction the initial yield of H_2 was 1.25 molecules/100 ev, whereas the initial yield with fast electron and gamma irradiation was 0.08 molecules H_2 /100 ev. The difference between these two yields should be the production of H_2 caused by the n, α reaction.

The dosages achieved during these irradiation studies represent about 100 seconds exposure in the ATR. Since the residence time in the ATR of the circulating control solution is expected to be less than 5 seconds per pass, the research data are equivalent to twenty or more passes through the reactor. Therefore, it should be easy to develop an engineering design for the system which will achieve the desired control of the gas content and pH of the solution.

It is interesting to note in the case of reactor irradiation that the initial yield of H_2 approaches the theoretical amount based on the standard energy absorption of 32 ev/ion pair. The low production of H_2 resulting from fast-electron and gamma irradiation emphasizes that their loss of energy is chiefly by processes other than by ionization.

8.353 Adjustment of Solution Concentrations^{1,2}

If sufficiently concentrated solutions of boric acid can be employed which make possible the use of natural boron, the removal of boron from the control solution can be accomplished by two methods, 1) a bleed and feed process which reduces the concentration by physical dilution, and 2) the ion-exchange removal from the circulating solution. At high initial concentrations, the removal by a chemical process like ion-exchange is neither practical nor economical, while a bleed and feed process soon reaches a point of diminishing returns for the effort expended at low concentrations. It can be shown however that the two methods used in tandem are advantageous for this operation. The actual physical dilution is straightforward. Concerning the ion-exchange process either Li-OH or H-OH (and probably NH_4 -OH) resin can be used for removal of boron from boric acid solutions.

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1. G. P. Simon, "The Removal of Potassium Tetraborate From Solutions by Ion-Exchange", WAPD-M(CDD)-118, February 1960.
 2. A. Squire, "Recommendation of a Chemical Shutdown Poison for all Naval Nuclear Plants", WAPD-M-413, May 10, 1960.

If (in order to achieve the B^{10} concentrations needed) it should be necessary to employ a boric acid control solution consisting of B^{10} enriched boron, the adjustment of solution concentrations and the recovery of the B^{10} boric acid will be accomplished as discussed in Section 4.4.

8.4 ETR Critical Facility, Reactor Physics Measurements

It was recognized that four 6 in. x 6 in. experimental spaces, surrounded with ETR fuel elements or control rods placed in the corners of the ETRC core and coupled with eight more fuel elements, making a fifth 6 in. x 6 in. experimental space, would approximate the geometry of the four-lobe ATR with 3 in. element thickness and a beryllium reflector. After preliminary reactivity measurements in the ETRC showed that an approximate loading could be made critical with two additional fuel elements, it was decided that critical experiments would be performed. The ideal geometry is shown in Figures 8.4E-I (Core I).

Because the program being carried out in support of ETR by the ETRC is quite pressing, only one week of ETRC time was allotted for the measurements reported herein. Therefore, it was not possible to make all the studies desired. Also, because time was limited, no modifications were made to the ETRC. With the exception of two experiment mockups, all materials used, e.g., fuel elements and control rods, were standard ETR or ETRC components.

8.41 Description of the Facility and Components

The ETRC is a low power, highly enriched, water cooled and moderated, beryllium-beryllium oxide reflected reactor. It is a full scale nuclear mockup of the core and reflector of its parent reactor, the ETR. In fact it is more than just a mockup because most of the inpile components (fuel elements and control rods) are identical to those in the parent reactor. For reasons of economy there are, however, slight modifications in the reflector. Over a period of several years, experiments have displaced enough 3 in. square beryllium pieces from the MTR to provide a complete reflector for the ETRC. These, however, make up only a 3 in. reflector as compared to 4-1/2 in. in the ETR. The remaining 1-1/2 in. are made up of canned beryllium oxide. Around the beryllium-beryllium oxide reflector is an aluminum reflector. The physical arrangements of the standard ETRC core and reflector are shown in Fig. 8.4A. The physical arrangement of the core, its support structure, control bridge, etc., are shown in Fig. 8.4B. A more detailed description of the facility can be found in IDO-16332¹.

With the exception of the experiment mockups, the core components (fuel elements, control rod fuel sections and guide tubes, and aluminum filler pieces) are standard ETR or ETRC components. The ETRC filler pieces are made of aluminum and have machined slots. These slots simulate the coolant passages in ETR filler pieces. The water volume associated with these pieces is 20 per cent of the total volume (metal-water ratio = 4). In all loadings studied an aluminum filler piece was also desired in grid position I-9 where a control rod guide

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1. D. R. deBoisblanc, et al., "The Engineering Test Reactor Critical Facility Hazards Summary Report", Phillips Petroleum Company, Idaho Falls, Idaho, Report IDO-16332, March 27, 1957.

tube is located. Aluminum plates were inserted in this guide tube to displace enough water to duplicate the metal-water ratio of the standard filler pieces.

The fuel elements which were used in these measurements contained 320 g U-235 and on the average 1.7 g natural boron. In addition, six elements with the same U-235 content but which contained no boron were used to shim the reactor by loading them with boron impregnated polyethylene tapes. The control rod fuel sections used contained 130 g U-235 and on the average 0.68 g natural boron. Drawings of the fuel element, and control rod fuel section and its guide tube are shown in Figures 8.4C and 8.4D.

The experiment mockups were made from stock aluminum rod (2.5 in. diameter) and schedule 40 stainless steel pipe (3.06 in. i.d. and 3.5 in. o.d.).

8.42 Loading Configurations

Inasmuch as no control rods are currently planned in the fuel region for the ATR it would have been desirable to mockup the core completely with fuel elements in the arrangement shown in Figures 8.4E-I. This was impossible in these critical experiments, however, because the safety rods were necessary for shutdown and because time did not permit removing the control rod guide tubes to permit insertion of fuel elements in those control rod positions which did not contain safety rods. The fuel arrangement desired intersected ten control rod positions. Fortunately, all four safety rod positions (H-7, K-9, G-11, M-11) and a driven shim (gray) rod position (H-13) were located in fueled regions. The remaining five positions contained fixed control rod fuel sections. As was mentioned previously, the other six guide tubes remained in the core but were water filled. These six tubes are not shown in the figures.

Because of these considerations, plus the boron built into the ETR elements, it was not possible to achieve criticality with the desired loading configuration. As a result, two extra elements had to be added. Because adding these elements produced flux distributions different from those in the desired loading, two loadings in which the two elements were placed in different locations were studied to determine the effects of these elements. In one loading the elements were located in the center experimental space so that almost symmetrical neutron flux distributions would be produced. In the second, the elements were located in the experimental space in diametrically opposite lobes. These loading arrangements are shown in Figures 8.4E-II and -III respectively. The effects on the neutron flux of removing the beryllium reflector from one of the lobes were also studied. The loading used for this study was the same as that in Fig. 8.4E-II, except that one-fourth of the reflector was removed. This arrangement is shown as a separate loading in Fig. 8.4E-IV.

8.43 Neutron Flux and Power Distribution

Thermal neutron flux distributions were measured to determine 1) the power densities required to obtain a thermal flux of 10^{15} n/cm²-sec in a hypothetical loop experiment, 2) the flux gradients that could be expected from narrow fueled regions, and 3) the effects of the reflector on the flux in an experiment located within a lobe. Most of the detailed measurements were made in and around the NE* quadrant, or lobe, which contained no control rod fuel sections. In fuel elements where detailed measurements were not made, only one measurement was made at the mid-plane in the center of the element. As will be seen in detail flux plots, this is the minimum flux within the element at the horizontal midplane.

8.431 Experimental Procedures and Treatment of Data

The thermal neutron flux distributions were determined from the activity of irradiated gold foils, and bare and cadmium covered gold wires. The foils are 5/32 in. diameter by 0.005 in. thick. The cadmium ratios were determined from 0.040 in. diameter gold wires, 3 in. long, the centers of which were covered with cadmium sleeves (0.020 in. wall and 1 in. long). After irradiation, the wires were cut into 1/4 in. sections and the average activity of the bare and covered sections were used to calculate the cadmium ratios. The cadmium ratios from sleeves have been normalized against standard cadmium covered foils.

The foils and wires, with cadmium sleeves, were taped with plastic electricians tape to either lucite strips, aluminum strips, or directly on an experiment. The lucite strips were used where measurements were made within a fuel element. These strips were inserted in the fuel element coolant channels and were designed to position the neutron detectors within ± 0.03 in. of the desired location. The aluminum strips were used where measurements were desired in a water space. These strips, containing the detectors, were taped on some object, usually a fuel element, such that they protruded into the desired region.

The activity of all the foils was first converted to relative flux distribution by use of the cadmium ratios and was then normalized to a Maxwellian-averaged flux of 10^{15} n/cm²-sec in the experiment mockup in the NE lobe. In some of the figures, a given value (usually the minimum) is normalized to unity so that the flux gradient is more easily determined.

8.432 Results

The pertinent neutron flux data which were measured in these experiments are presented in Figures 8.4F through 8.4N.

* The top of the page is assumed to be North in this description. This conforms to the physical orientation of ETR.

It can be seen in Figures 8.4F and 8.4G that maximum-to-minimum flux in the NE lobe is not appreciably affected by the rearrangement of elements from Loading II to Loading III but is significantly increased by the removal of the beryllium reflector as is evidenced in Fig. 8.4H. It is significant, however, that even though there is a pronounced gradient across the lobe, the maximum-to-average flux around the aluminum cylinder is only 1.05 in Loading II changes to only 1.10 in Loading IV, which has a quite different power distribution. This is illustrated in a circumferential flux plot in Fig. 8.4I. An E-W traverse through the lobe is shown in Fig. 8.4J. This property has been recognized as a most important feature of ATR and is discussed more completely in Section 3.2.

It can be seen in the detailed flux traverses in Figures 8.4K and 8.4L that undesirable flux gradients exist within the fuel elements. For example, the maximum-to-minimum flux within the fuel element in position N-6 is 1.9. This indicates the desirability of graduating the fuel density and increasing the absorption cross section of the fuel element side plates by the addition of burnable poison or possibly fuel.

A typical vertical traverse, shown in Fig. 8.4M, has approximately the same shape and maximum-to-average flux (1.40) as traverses measured in ETR loadings.

A flux traverse measured in grid positions H-10, J-10, and K-10 is shown in Fig. 8.4N merely to indicate the amount of flux peaking in a 3 in. x 3 in. space containing an aluminum filler piece with a metal-to-water ratio of four.

8.433 Power Calculations

Various factors associated with power distributions have been calculated and are summarized in Table 8.4A. Total power is shown for the most balanced loading only since the other cores are not considered representative of the manner in which ATR will be loaded. Of considerable significance is the fact that even though the maximum-to-average in the NE lobe varied from 1.54 to 2.35, the power in that lobe required to produce the same experimental flux changed only 10 per cent.

To make these power calculations, special considerations were required. The thermal neutron flux values shown in the diagrams of the three experimental loadings (Figures 8.4F, 8.4G, and 8.4H) were all measured at the center of the fuel elements and normalized to a thermal flux of 1×10^{15} n/cm²-sec at the center of the mockup loop in the NE lobe. For power calculations each value had to be corrected to an average thermal flux per element. Although the maximum-to-average ratios of the vertical flux profiles are essentially constant throughout the core, this is not true of the individual (within each element) horizontal profiles, as shown in Figures 8.4K and 8.4L. For this reason, the correction factor used to obtain an average flux from the measured value at the center of the element depended upon the particular location of the element in the core. These factors (over-all center-to-average) ranged from 0.77 to 0.98.

TABLE 8.4A
POWER DISTRIBUTIONS

	NE LOBE*			FULL CORE
	Loading II	Loading III	Loading IV	Loading II
Total Power	72 Mw	77 Mw	70 Mw	335 Mw (425)**
Horizontal Max/Avg Power	1.62	1.54	2.35	2.61

* 12 Elements: K-5 to 8, N-5 to 8, L-5 to 8, and M-5 to 8.

** Power of core loading II corrected to 4 ft length.

In calculating power, an empirically determined correction factor was included to account for the contribution of epithermal neutrons to the total fission rate. The fission rate due to epithermal neutrons is 10 per cent of that due to thermal neutrons. The U-235 fission cross section at 2200 meters/sec (580 barns) was corrected by the usual factors to a Maxwellian-averaged value at 75°F. Using this cross section, the measured flux ratios, and correcting to a core length of 4 ft gave a power level of 425 Mw. In Section 3.2 a calculation of a 3 in. four-lobe reactor similar to ATR gave a value of 220 Mw. Even with allowance for the difference in neutron temperature (75°F in the ETRC, 200°F in the ATR) and computational uncertainties, the advantage of the more symmetric geometry is evident.

8.44 Critical Mass - Measurements and Calculations

It was mentioned above that the four-lobe geometry shown in Fig. 8.4E-I (Core I) needed two additional fuel elements in the center of the core for criticality (positions J-9 and H-10). This loading (Core II, Fig. 8.4E-II) was critical with the driven shim rod in position H-13 at 17.5 in. (full travel 36 in.). All other rods were fully withdrawn, i.e., the fuel sections on all other rods were in the core.

Forty-two of the forty-eight fuel elements in Core II were standard ETR fuel elements with an average of 1.72 g of natural boron (ranging from 1.60 to 1.86 g). The remaining six elements, like the former in all other respects, were fabricated without boron. Since the

shim rod was worth only 0.3 per cent $\Delta k/k$, the coarse shimming was done by adding boron impregnated polyethylene tapes to the six unborated elements. The U-235 mass in this loading (48 fuel elements and 9-1/2 control rod fuel sections) is 16,600 g. The critical mass of the cold clean unborated loading with the configuration of Core I was derived by calculating the excess reactivity this configuration would have had if there had been no boron in the fuel elements. This excess reactivity is then used to calculate the amount of fuel which would have to be removed uniformly from Core I (46 elements, 10 control rod fuel sections - 16,020 g) to reduce its excess reactivity to zero, or in other words, k_{eff} to unity. Table 8.4B is an itemized list of the results of these calculations using a theoretical approach in one case and an empirical approach in another. As shown, the cold clean unborated critical masses by the two methods are in good agreement.

TABLE 8.4B
CRITICAL MASS CALCULATIONS

	Empirical	Theoretical
Removing boron	+ 0.0906 $\Delta k/k$	+ 0.0860 $\Delta k/k$
Removing extra elements	- 0.0445	- 0.0355
Complete withdrawal of shim rod	+ 0.0016	+ 0.0016 (same)
Total excess reactivity	+ 0.0477 $\Delta k/k$	+ 0.0521 $\Delta k/k$
Excess fuel	2760 g	3340 g
Critical mass (16,020 - Excess fuel)	13,260 g	12,680 g

FIG. 8.4A
ETRC CORE DIAGRAM

P P CO-C-2751

FIG. 8.4B
ETR CRITICAL FACILITY
PERSPECTIVE

PP CO -D-2752

FIG. 8.4C
ETR AND ETRC FUEL ASSEMBLY

P.P. CO. - C - 2753

ETR AND ETRC CONTROL ROD
FUEL SECTION AND GUIDE TUBE

FIG. I

FIG. II

FIG. III

FIG. IV

CR — CONTROL ROD

— ALUMINUM FILLER PIECE

S.S. — EXPERIMENT MOCKUP
ALUM.

— FUEL

— BERYLLIUM REFLECTOR — H₂O

— BeO

FIG. 8.4E
MOCKUP LOADING IN THE ETRC

P.P. CO-C-2755

PP CO-A-2757

FIG. 8.4F
LOADING II MIDPLANE THERMAL NEUTRON FLUX, 10^{14} n/cm²-sec.

P.P.CO-A-2758

FIG. 8.4 G
LOADING III MIDPLANE THERMAL NEUTRON FLUX, 10^{14} n/cm² - sec.

FIG. 8.4H
LOADING IV MIDPLANE THERMAL NEUTRON FLUX, 10^{14} n/cm²-sec.

FIG. 8.4 I
EXPERIMENT CIRCUMFERENTIAL THERMAL NEUTRON FLUX

PP CO-A-2760

FIG. 8.4 L
FUEL ELEMENT THERMAL NEUTRON FLUX TRAVERSES

FIG. 8.4M

TYPICAL VERTICAL NEUTRON FLUX DISTRIBUTION

FIG. 8.4 N

HORIZONTAL THERMAL NEUTRON FLUX TRAVERSE
LOADING II

8.5 Two-Dimensional Computation of Four-Lobe ETR Geometry

A four-group, two-dimensional calculation was made duplicating ETR Critical Facility Loading II described in Section 8.4. The IBM 704 code PDQ02¹ was utilized for this calculation.

Loading II went critical with the shim rod in position H-13 at 17.5 in.; however, for the two-dimensional calculation this rod was assumed to be fully withdrawn and the position filled with the 130 g shim fuel section. It was necessary to extend and deform the volumes of the "Experiment Mockup" in the NE and SW lobes for the calculation in order to fit the available mesh spacing. A schematic diagram of the core is shown in Figure 8.5A. Aluminum guide tubes were present in Loading II in positions E-9, G-9, J-5, J-7, and M-9. These tubes are not shown in Figure 8.5A but the calculation included these tube regions.

8.51 Nuclear Constants

The nuclear constants which were used are listed in Table 8.5A. The constants for the fast groups (1, 2, and 3) were obtained by using the IBM 650 code MUFT III. The thermal constants (Group 4) are Maxwellian-averaged values at 20°C.

8.52 Critical Mass

The eigenvalue obtained from the PDQ calculation was 1.037 indicating a supercritical core. The "critical mass" corresponding to an eigenvalue of 1.000 can be approximated by using the following equation.

$$\frac{\Delta K}{K} = (1 - f) \frac{\Delta M}{M},$$

where

$$\frac{\Delta K}{K} = 0.037,$$

thermal utilization, f , = 0.75, and

total U-235 mass, M , = 16.66 kg.

ΔM is found to be 2.46 kg, corresponding to a computed "critical mass" for this core of 14.2 kg of U-235. This compares to the measured value of 16.6 kg as given in Section 8.44.

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1. G. G. Bilodeau, W. R. Cadwell, J. P. Dorsey, J. G. Fairey and R. S. Varga, "PDQ-An IBM 704 Code to Solve the Two-Dimensional Few-Group Neutron-Diffusion Equation", WAPD-TM-70, August, 1957.

TABLE 8.5A

NUCLEAR CONSTANTS USED IN ETRC-PDQ CALCULATION

Composition		Group 1	Group 2	Group 3	Group 4
Water	D	2.2895	1.10486	0.59419	0.161
	Σ_a	0.001415	0.000012	0.000942	0.01911
	Σ_R	0.1036	0.14941	0.15022	---
Aluminum	D	24.0	6.76	4.55	4.00
	Σ_a	0	0	0.00017	0.0139
	Σ_R	0.00181	0.00181	0.000403	---
BeO	D	2.6832	0.6861	0.621232	0.552
	Σ_a	0.001862	0	0	0.000510
	Σ_R	0.03057	0.019195	0.010634	---
Beryllium	D	3.30	0.59764	0.500	0.411
	Σ_a	0.000028	0	0.000021	0.0015
	Σ_R	0.03324	0.031737	0.01938	---
320 g fuel element with 1.7 g of boron in element	D	3.1145	1.32939	0.76278	1.27061
	Σ_a	0.001042	0.000383	0.00644	0.1158
	Σ_R	0.06439	0.090134	0.087554	---
	$\nu\Sigma_f$	0.000548	0.000715	0.009446	0.182
320 g fuel element with 0.8 g boron present in polyethylene strips	D	3.1151	1.3296	0.763	1.30055
	Σ_a	0.001041	0.000379	0.006357	0.1096
	Σ_R	0.06439	0.09013	0.08775	---
	$\nu\Sigma_f$	0.000548	0.000715	0.00945	0.182
320 g fuel element with 0.4 g boron present in polyethylene strips	D	3.11537	1.32972	0.763105	1.3144
	Σ_a	0.001041	0.000377	0.00623	0.1069
	Σ_R	0.06439	0.09013	0.087833	---
	$\nu\Sigma_f$	0.000548	0.000715	0.009452	0.182
130 g shim rod fuel section	D	3.70843	1.48852	0.88673	0.32323
	Σ_a	0.000697	0.000158	0.002963	0.05676
	Σ_R	0.0493	0.068	0.067	---
	$\nu\Sigma_f$	0.000222	0.00029	0.003851	0.07206
Experiment mockup	D	2.792	1.3283	0.73716	0.40903
	Σ_a	0.000307	0.000015	0.002806	0.06385
	Σ_R	0.04559	0.03962	0.03745	---
Aluminum filler (Al + 20% H ₂ O)	D	5.2365	1.72671	1.14286	0.650
	Σ_a	0.000285	0.000002	0.000333	0.01385
	Σ_R	0.0236	0.03226	0.0322	---
Water plus aluminum guide tube	D	2.63182	1.20199	0.66664	0.19933
	Σ_a	0.001108	0.000009	0.000792	0.01813
	Σ_R	0.084	0.1195	0.120	---

8.53 Neutron Flux

To compare the PDQ values of thermal neutron flux distribution with the measured distribution, the PDQ values were normalized by multiplying them by the ratio of the measured thermal neutron flux in position K-8 to the corresponding PDQ value. The calculated values of thermal neutron flux are shown in Fig. 8.5A.

8.54 Reactor Power

The total reactor power was calculated from the formula:

$$\text{power} = C \sum_{i=1}^4 \int V \Sigma_{fi} \Phi_i dV ,$$

where

Σ_{fi} = the macroscopic fission cross section for neutron energy group i as taken from Table 8.5A. v was assumed to be equal to 2.47 for all groups,

Φ_i = neutron flux in energy group i normalized to measured values by the thermal group ratio in position K-8, and

C = conversion factor to convert fissions/sec to power.

Using the above formula and constants, the reactor power corresponding to the thermal neutron fluxes given in Fig. 8.5A was calculated to be 330 Mw. A value of 335 Mw as listed in Table 8.4A was obtained from the measured neutron fluxes. In a similar manner the power generated in the NE lobe was calculated to be 73 Mw, compared with 72 Mw listed in Table 8.4A. The power generated in several fuel elements including those around the NE lobe was also calculated and is listed in Table 8.5B along with values obtained in ETRC Loading II. Also listed in Table 8.5B are calculated values of power density at the center of several fuel elements and the ratio of power produced by above thermal fissions to the power produced by thermal fissions at the center of the elements.

TABLE 8.5B

COMPARISON OF CALCULATED AND MEASURED ETRCFUEL ELEMENT POWERS

Fuel Element Position	Power Generated Within Element, Mw		Power Density at Center of Element Mw/liter	Ratio - $\frac{\text{Above Thermal Fissions}}{\text{Thermal Fissions}}$
	PDQ Calculation	ETRC Measured		
E-8	4.81		1.12	0.0564
F-8	7.00		1.63	0.0517
G-8	9.08		2.15	0.0554
H-8	9.56		2.35	0.0678
I-8	11.18		2.76	0.0627
J-8	10.75		2.64	0.0672
K-8	9.39	9.81	2.31	0.0713
L-8	9.18	9.51	2.18	0.0559
M-8	7.00	6.29	1.63	0.0533
N-8	5.05	4.55	1.18	0.0560
K-7	9.23	8.97		
K-6	7.31	6.72		
K-5	5.19	4.52		
L-5	4.72	5.01		
M-5	4.12	4.68		
N-5	3.34	3.75		
N-6	4.08	4.36		
N-7	4.62	4.41

FIG. 8.5 A

PPCO-A-2766

MIDPLANE THERMAL NEUTRON FLUX AT A REACTOR POWER OF 330 MEGAWATTS (UNITS OF 10^{14} NEUTRONS / cm^2 - sec)

8.6 Fuel Element Supplement

8.61 Other Designs Considered

8.611 Rectangular Elements

Rectangular fuel elements were considered for use in the ATR. These elements have the advantage of easy construction and interchangeability between the concave and convex fuel annuli. However, the wasted space between the fuel elements which results from fitting rectangular elements into annular segments reduces the over-all core efficiency and thereby negates the advantages of the simple construction. If rectangular fuel elements of the MTR type were specified for the ATR, they would be required to have a local power density approximately 24 per cent greater than that required for the equivalent circular segment fuel element. Also, those segments of the fuel annulus which would contain no fuel because of the use of rectangular fuel elements would probably cause an undesirable shifting of the neutron flux distribution.

The rectangular elements are ideally suited for a square-lobe configuration such as mocked up in the ETRC (Fig. 8.4E). However, this configuration is not recommended for the ATR because of the high total power and high maximum power density required. (Re: Table 8.4A).

8.612 Tubular Elements

Bundles of tubes in various arrangements were also considered for the reactor core. If it were possible to obtain adequate strength in these tube bundles without the use of side plates or exterior shells, then the tube bundles would be superior to the equivalent plate designs. However, tube bundles of the required length and diameter, fabricated by existing techniques, do not have sufficient strength to withstand the hydraulic forces without side plates or shells. Once these shells are added, the tube bundles have no superiority over equivalent plate designs, and the added complexity of the tubes makes them less desirable than the plate type element.

8.613 Involute Elements

One of the most promising type of fuel elements considered for uniformly loaded annular fuel elements was the involute element of a design similar to that proposed by Internuclear Company, Inc. (Report AECU-3775). However, the involute element could not be adapted for efficient use in the ATR core because of the reversal of the direction of curvature of the fuel elements within the core.

8.62 Discussion of Materials Considered

8.621 Fuel Cladding and Side Plates

Aluminum alloys have been recommended for fuel cladding and side plates because of their low neutron cross section, availability,

low cost, ease of fabrication, and adequate mechanical and chemical properties.

The corrosion resistance of aluminum to high temperature water is determined primarily by the purity and the metallurgical condition of the metal. High temperature heat treatment seriously impairs resistance and leads to intergranular attack, whereas cold work is beneficial. The presence of iron and silicon either as impurity or alloy additions improves the resistance of aluminum to corrosion below 400°F. Above this temperature intergranular penetration occurs although such elements as nickel, iron, titanium, copper, cobalt, molybdenum, or tungsten may be added to eliminate this; nickel being the most effective. Al-Fe-Ni type alloys have been developed which show very good corrosion resistance to water at temperatures up to 680°F.

Since the maximum temperature of the fuel plates under operating conditions is in the range of rapid corrosion of 1100 aluminum alloy and in the range of high strength loss for the 6061 age-hardened alloy, an X8001 aluminum alloy containing 1 per cent nickel is recommended. The corrosion of this alloy changes from localized penetration at grain boundaries and blister formation to the more desirable uniform overall attack. The corrosion of X8001 alloy has been studied by various groups with the rate being about 0.2 mil/month under static conditions. Flow velocities up to 20 ft/sec show an increased rate of about three fold. Corrosion of aluminum alloys during higher flow velocities and increased temperatures should be incorporated in a research and development program.

Increased corrosion resistance can be obtained by adding small amounts of Ti and Be to X8001 and controlling Si content, or by using X8002 or X8003 aluminum alloys. Under aqueous static conditions at 600°F, the corrosion resistance of X8002 and X8003 alloys are respectively five-fold and sixteen-fold that of X8001 alloy. The X8003 alloy would have the disadvantage of being more expensive because of the higher purity of the base material used.

Aluminum alloys having improved mechanical properties over the 1100 alloy at high temperatures must be used to increase the strength of aluminum clad fuel plates for high operating temperatures. Table 8.6A compares the strength of several aluminum alloys at various temperatures. Although other aluminum alloys with good high temperature strengths might have been included in Table 8.6A, the good corrosion resistance of the alloys shown makes them logical choices for the ATR.

8.622 Fuels

Two types of metallurgical structures for the fuel material have been considered, uranium-aluminum alloys and uranium oxide dispersed in an aluminum matrix. When uranium-aluminum alloys are prepared containing greater than 13 weight per cent uranium, some fabrication difficulties may be encountered. These hypereutectic alloys are strong and brittle due to the intermetallic compound UAl_4 . Rapid cooling of these high uranium alloys tends to suppress the peritectic reaction

TABLE 8.6A
STRENGTHS OF ALUMINUM ALLOYS^{1,2}

Alloys	Yield Strength, psi				
	75°F	300°F	400°F	500°F	600°F
1100 Al					
"O" Condition	5000	4500	3500	2000	1500
H-18 Condition	22,000	14,000	4000	2000	1500
X8001					
Forged (F)	17,800	13,700	9400	6000	3200
X8002 H-18 Condition	28,800		12,000	7200	5600
6061 Al					
T-6 Condition	40,000	29,000	15,000	5000	2500
M257 Extruded (F)	24,000		18,000	16,000	15,000
M430 Extruded (F)	35,000		30,000	27,000	23,000

1. "Alcoa Aluminum Handbook", Aluminum Company of America, Pittsburgh, Pennsylvania, 1959.
2. W. W. Binger, Alcoa--personal communication.

occurring at 750°C and prevent UAl_3 from reacting with Al to produce UAl_4 . Suppressing the peritectic reaction produces a more ductile material which makes it possible to hot roll U-Al alloys containing up to 35 weight per cent uranium. Uranium-aluminum alloys containing greater than 35 weight per cent uranium can be fabricated by using ternary additions (Ge, Zr, Sn, Si) to suppress the peritectic reaction, thus allowing the primary nucleating compound UAl_3 to be retained as the stable phase. Experimental studies have shown that a 48 weight per cent U-Al alloy containing 3 weight per cent silicon can be successfully hot rolled. Because of the difficulties encountered in reprocessing uranium in the presence of silicon, zirconium, tin or germanium appear more desirable ternary additives.

A 35 weight per cent U-Al alloy contains approximately 52 weight per cent of the intermetallic compound UAl_4 unless some suppression is obtained. Without any suppression of the UAl_4 compound, the high yield strength of the core alloy and the low strength of the cladding produces "dog boning". Such nonuniform deformation is not pronounced in plates containing 10 to 20 weight per cent U-Al alloys because of close matching of the yield strength and ductility of these alloys with 1100 Al containment materials at rolling temperature. Laboratory tests have shown that "dog boning" can be reduced to the point where it is no longer a problem in alloys containing over 35 weight per cent uranium if an aluminum alloy is used for the picture frame whose high temperature properties are similar to that of the core.

If fuel loadings requiring higher uranium content are necessary, the plates can be fabricated using UO_2 -Al dispersion cores containing up to 60 weight per cent UO_2 . Highly enriched UO_2 costs more than the metal, and rejected plates prepared by powder techniques must be chemically recovered rather than recycled directly into the melt as is the case with the alloy. Although the powder metallurgy plates are more costly, they are superior to the alloy because of less dog boning, and burnable poisons can be added with much greater accuracy and greater uniformity than with the alloys. Also, gradients in fuel concentration can be easily employed if desired. Fuel plates containing fuel gradations as large as 30 weight per cent have been produced by the powder technique. Other advantages which might be considered are that fewer steps are employed in preparation of plates and that less scrap is formed in the processing of plates.

8.63 Extension to Higher Power Operation

From a heat removal standpoint, reactor power could be increased, if desirable, in one or more of many ways. The easiest method of increasing reactor power would be to increase the coolant velocity. This would require no modifications to the core or to the element, except for possible modification in the total fuel loading. At the present state of reactor technology, 45 to 50 ft/sec appears to be the maximum practical coolant velocity for use with Al plates. The reactor tank pressure may also be increased, which would allow a corresponding increase in the maximum wall temperature. However, again current

technology for aluminum clad fuel plates limits the maximum, practical wall temperature to approximately 475° F and therefore little additional gain may be made by increasing the tank pressure above 300 psi without changing the cladding materials. APM alloys have been developed which at 700° F have mechanical properties better than 1100 Al at room temperature. The APM alloys containing iron and nickel have been corrosion tested in water at 680° F and show excellent performance at elevated temperatures. Additional testing of these materials is required, but preliminary tests indicate that aluminum technology could possibly be extended to include 700° F wall temperature.

The greatest potential increase in core power may be obtained through the use of variable fuel and/or poison loadings throughout the fuel elements. If the fuel and/or poison concentration is held constant in any given fuel plate, but is adjusted from plate to plate so that the maximum heat flux is the same for each plate, then the power of the ATR could be increased by a factor of approximately 1.5. Such a variation from plate-to-plate is entirely practical and may be accomplished through the use of standard fabrication techniques. If the fuel and/or poison is also varied lengthwise in each plate, the reactor power may be increased by an additional factor of approximately 1.3. However, the longitudinal variation within a single plate may cause considerable fabrication difficulties which would necessitate a development program and cause an increase in the cost of the element. In any event, deviation from uniform fuel loadings would cause some variation of the reactor flux, and for each proposed case of variable loadings, flux calculations would be necessary to determine the detailed effect of the variable fuel and/or poison loadings.

8.7 Reactor Cooling

8.71 Introduction

The hydraulic and heat transfer performance of several different fuel element geometries was evaluated during this study. This work was undertaken to facilitate the selection of a fuel element which is expected to operate satisfactorily at the specified conditions.

The general criteria used to establish the fuel element geometry and operating conditions of the primary coolant are discussed in Section 4.3, 4.5, and 8.6. The physics requirements are discussed in Section 3.0 and the core configuration which the fuel element must fit is discussed in Section 2.0.

Standard, non-isothermal, turbulent hydraulic calculations have been made to determine the core pressure drop and the absolute pressure at the hot spot.

8.72 Circular Segment Heat Transfer

An explanation of the method used to calculate the wall temperature is presented here since it differs slightly from that sometimes used. The nominal wall temperature at the point of maximum power density is calculated from the following equation.

$$T_w = T_{in} + \Delta T_b + \Delta T_f ,$$

or

$$T_w = T_{in} + \left(\frac{\phi_{max}}{\phi_{avg}} \right)_H \frac{(Q/A)A'}{V C_p \rho \alpha} + 0.90 \left(\frac{\phi_{max}}{\phi_{avg}} \right)_H \left(\frac{\phi_{max}}{\phi_{avg}} \right)_V \frac{(Q/A)}{h} ,$$

where

T_w = wall temperature,

T_{in} = inlet temperature,

ΔT_b = bulk water temperature rise to the point in question,

(Q/A) = total reactor power divided by total reactor heat transfer area,

A' = heat transfer area in hot channel from top of channel to hot spot.

V = coolant velocity,

ρ = coolant density,

C_p = coolant heat capacity,

α = cross sectional flow area of hot channel,

$$\left(\frac{\Phi_{\max}}{\Phi_{\text{avg}}} \right)_H = \text{horizontal power density ratio, max/avg,}$$

$$\left(\frac{\Phi_{\max}}{\Phi_{\text{avg}}} \right)_V = \text{vertical power density ratio, max/avg, and}$$

h = heat transfer coefficient evaluated at a film temperature equal to $\frac{1}{2}(T_{\text{in}} + \Delta T_b + T_w)$.

To each of the terms in this equation is applied its maximum deviation for the hot-spot hot-channel analysis. These uncertainty terms are grouped together in $\Pi(F_b)$ and $\Pi(F_h)$. The heat transfer coefficient is now evaluated at the hot-spot film temperature, i.e.,

$$\frac{1}{2}(T_{\text{in}} + \Delta T_{\text{add}} + \Pi(F_b) \Delta T_b + T_{w_{\max}}).$$

The equation, containing the uncertainty factors, for the maximum wall temperature becomes:

$$T_{w_{\max}} = T_{\text{in}} + \Delta T_{\text{add}} + \Pi(F_b) \left(\frac{\Phi_{\max}}{\Phi_{\text{avg}}} \right)_H \frac{(Q/A)A'}{V \rho C_p \alpha} + 0.90 \Pi(F_h) \left(\frac{\Phi_{\max}}{\Phi_{\text{avg}}} \right)_V \left(\frac{\Phi_{\max}}{\Phi_{\text{avg}}} \right)_H \frac{(Q/A)}{h},$$

where

$T_{w_{\max}}$ = maximum creditable wall temperature,

ΔT_{add} = variation of inlet bulk water temperature,

$\Pi(F_b)$ = product of all hot-channel factors, and

$\Pi(F_h)$ = product of all hot-spot factors.

The hot-channel, hot-spot and power distribution factors used in the heat transfer analysis are cited in Table 4.5A. Most of these factors are based upon MTR-ETR experience. The power distribution factors are set by physics studies.

Several of the factors are set by the fuel assembly specifications. These are: fuel content per plate, fuel core alloy thickness, and fuel segment angle. In calculating the hot channel factor for the hydraulic diameter variation it was assumed that the velocity was directly proportional to the square root of the hydraulic diameter. The cross sectional area variation of a coolant channel must be included since it is necessary to convert from a measured, or calculated, linear velocity to a mass rate of flow.

The net heat generation factor (0.90) results from the fact that not all of the reactor heat is generated in the fuel core alloy.

The modified Colburn equation was used to calculate the heat transfer coefficient;

$$\frac{hD}{k} = 0.023 \text{ Re}^{0.8} \text{ Pr}^{0.3},$$

where

h = heat transfer coefficient, Btu/hr-ft²-°F,

D = hydraulic diameter, ft,

k = thermal conductivity, Btu/hr-ft²-°F/ft,

Re = Reynolds number, and

Pr = Prandtl modulus.

All the physical properties were evaluated at the film temperature. The coefficient calculated by this equation has an uncertainty of ± 20 per cent or a hot-spot factor of 1.25 to account for the variation in the experimental measurements and also since recent investigations have shown that the modified Colburn equation predicts heat transfer coefficients which are larger than those that were measured in geometries similar to those studied here. Table 8.7A and Figures 8.7A and 8.7B summarize the results of the calculations for the selected geometry and inlet coolant conditions. Changing the channel size from the 0.078 in. value does not affect the operable power significantly if the core pressure drop and maximum wall temperature are held constant. Increasing the length of the core from 4 ft to 6 ft decreases the allowable power density by about 1 Mw/1 for the 0.078 in. channel case. Table 8.7B cites the lateral differential pressures that were calculated for various flow rates in the 0.078 in. channel. It appears that the element selected can be expected to withstand the hydraulic forces developed during normal operation.

TABLE 8.7A

ATR CIRCULAR-SEGMENT FUEL ELEMENT HEAT TRANSFER AND HYDRAULICS SUMMARY*

Coolant Vel. ft/sec	$(Q/A)_{avg}^{**}$ Btu/hr ft ²	Maximum Allowable Core Power Mw	Average** Power Density (Mw/liter)	Maximum Power Density (Mw/liter)	T _{wall} (°F)	Flow Thru Fuel Core (gpm)	Core Δp (psi)
30	4.50×10^5	189	0.72	1.89	432	17,300	46
40	5.67×10^5	238	0.91	2.38	424	23,100	76
45	6.14×10^5	257	0.98	2.57	418	26,000	95
50	6.64×10^5	278	1.06	2.78	414	28,900	115

* Experimental Conditions

Fuel element sector	= 45°
Fuel element thickness	= 2.5 in.
Number of plates	= 19
Thickness of plates	= 0.050 in.
Thickness of channels	= 0.078 in.
Inlet water temperature	= 130°F
Inlet water pressure	= 300 psig
Metal-to-water ratio	= 0.80
$(\Phi_{max}/\Phi_{avg})_V$	= 1.4
$(\Phi_{max}/\Phi_{avg})_H$	= 1.87

** Average over entire core.

TABLE 8.7B

LATERAL DIFFERENTIAL PRESSURES FOR VARIOUS FLOW RATES

Velocity ft/sec	Lateral Differential Pressure psi
30	2.1
40	3.8
45	4.8
50	5.9

FIG. 8.7 A
 ALLOWABLE POWER DENSITY VERSUS COOLANT VELOCITY

FIG. 8.7 B
 TOTAL ALLOWABLE REACTOR POWER VERSUS COOLANT VELOCITY

8.8 Materials of Construction

Requirements for high neutron flux and test loop pressure can result in relatively high combined stresses in the reactor vessel components and the in-pile loop tubes. In the design of the four-lobe reactor wherein the vessel wall is removed from the vicinity of the fuel so that the internal heating rate is low (<0.4 watt/g) stainless steel (Type 304L or 347) is recommended as the pressure vessel material.

The inner wall of the D₂O reflector tank (See Fig. 2.0A) is in a region of high internal heat generation (10 watts/g) and can be designed to meet the wall thickness requirement of the ASME Code and of that desired by nuclear considerations only for certain design conditions (See Section 8.85). A cylindrical inner tank of aluminum alloy 5154-0 can meet ASME Code for these specified maximum design conditions.

For the alternate arrangement with a beryllium reflector (Fig. 2.0B and Section 8.86) a service lifetime of the beryllium in excess of two years can be predicted for a design wherein other metals in the cusp regions reduce the maximum exposure.

The specifications for decontamination of the in-pile loops necessitate the use of stainless steel (Type 304L or 347) as the construction material. These loops are designed for a definite fatigue life of 65,000 cycles according to tentative design criteria for the maximum design conditions of pressure, cycles of operation, and combined stress.

8.81 Internal Heating Rate

Gamma heating rates in various components of the ATR were calculated¹ based on an operating power density of 1.0 Mw/liter. Profiles of the average gamma heating along radial directions between two of the lobes and through one of the lobes were presented. All values are average axial values so that centerline values are obtained by multiplying by the axial maximum-to-average power ratio given in Table 8.7A as 1.40. The contribution from neutron heating to the total heating due to elastic and inelastic scattering and capture is approximately 5 per cent but varies (from n , γ reactions, etc.) according to the composition of the material.

Values of the internal heating rates used in the stress calculations are as follows:

	Total Heat Generation Watts/g
In-pile tubes	20
D ₂ O inner tank wall	10
D ₂ O outer tank wall	0.11
Pressure vessel	0.02

1. Internuclear Company, Clayton, Missouri, Technical Memorandum, TM-FAL-60-6, June 27, 1960.

8.82 Material Selection

With potentially high thermal stresses from internal heat generation and with high strength at temperature required, the combined stresses in various materials were compared with the maximum allowable stress values at temperature as required by the ASME "Boiler and Pressure Vessel Code".¹ For those designs in which the pressure vessel is inside the outer reflector so that the internal heat generation is greater than 2 watts/g the total stress in a stainless steel pressure vessel becomes greater than the allowable stress, while the total stress in an aluminum alloy vessel is less than the allowable stress. In the design wherein the vessel is outside the outer reflector so that attenuation of neutrons and gammas in the water and possibly the thermal shield occurs, steel as a material for the pressure vessel shell is more attractive. According to a prominent fabricator of aluminum shapes, the largest tubes extruded are 24 in. o.d. so that allowable stresses for vessel shells in excess of 24 in. according to ASME code would be based on the annealed value.

In Table 8.8A are thermal stress factors and ASME code allowable shell thickness for various materials at several tube diameters for the following design conditions:

Internal pressure	300 psig
External pressure	0 psig
Maximum metal temperature	200°F
Welding joint efficiency	80 per cent

It is noted that high strength aluminum alloys offer an advantage in reduced thermal stress by greater than a factor of 15.

The corrosion resistance of these alloys at 200°F should be adequate, although the possibility for stress corrosion does exist. The beneficial effects of copper additions to aluminum and especially of nickel on the corrosion resistance in high purity water at elevated temperatures have been noted by several authors.^{2,3}

8.83 Stresses in Reactor Vessel Components and Loop Tubes

The stresses in the loop tube and the reactor vessel components were studied to determine the thermal and pressure stresses. The methods of calculation are briefly described.

The property values assumed for the aluminum alloys and for stainless steel are temperature dependent. The values used in the calculations are given in Table 8.8B (for temperature range 77 - 212°F).

1. ASME Boiler and Pressure Vessel Code, 1959, Section VIII, Unfired Pressure Vessels.
2. R. L. Dillon, R. E. Wilson, W. H. Troutner, "High Temperature Aqueous Corrosion of Commercial Aluminum Alloys", HW-27636, January 11, 1956.
3. J. E. Draley and W. E. Ruther, "Corrosion Resistant Aluminum Above 200°C", ANL-5430, July 15, 1955.

TABLE 8.8A

THERMAL STRESS FACTORS AND ASME CODE

ALLOWABLE SHELL THICKNESS AT 200°F

Materials	Tube i.d. Inches	Thickness Inches	$\frac{\alpha E}{k(1-\nu)}$
Aluminum Alloys			
1100-0	39.0	3.53	1.62
5052-H34	39.0	0.90	2.62
6061-T6	39.0	0.84	2.25
5154-0	39.0	1.03	2.89
	40.5	1.07	2.89
	59.0	1.56	2.89
	72.0	1.90	2.89
	88.5	2.33	2.89
Stainless Steel			
Type 347	39.0	0.39	38
	59.0	0.60	38
	72	0.73	38
	88.5	0.90	38
Zirconium	-	-	10

TABLE 8.8B

PROPERTY VALUES OF VARIOUS ALLOYS USED INTHERMAL STRESS CALCULATIONS

Property Values	Zirconium	Aluminum 5154-0	Stainless Steel
Coefficient of thermal expansion, in./in.°F	4.0×10^{-6}	14.0×10^{-6}	9.0×10^{-6}
Modulus of elasticity, psi	13.7×10^6	10.3×10^6	28×10^6
Poisson's Ratio	0.33	0.33	0.29
Thermal conductivity, Btu/hr-ft-°F	8.2	74.5	9.4

8.831 Thermal Stress

The rate of internal heat generation will not be uniform through the metal due to attenuation so that the heat generation will be higher at the inner surface. Since cooling is also accomplished at the inner surface, the temperature distribution through the metal will be less than for equivalent uniform heat generation. A value of thermal stress leading to conservative design will therefore be obtained by an assumption of uniform internal heat generation.

The formula used assumes that the cylinder is free to expand in an axial direction, that the vessel is relatively long, and that the variation in temperature along the axis is gradual so that axial temperature differences do not cause appreciable stresses. The maximum tangential stress in a hollow tube is given by:

$$\sigma = \frac{\alpha E}{(1-\nu)} \left\{ \frac{q}{8k} \left[b^2 + a^2 - \left(\frac{b^2 - a^2}{\ln b/a} \right) \right] + (T_b - T_a) \left[\frac{a^2}{b^2 - a^2} - \frac{1}{\ln b^2/a^2} \right] \right\},$$

where

σ = the longitudinal and tangential stress, psi,
 α = coefficient of thermal expansion, in./in. $^{\circ}$ F,
 E = modulus of elasticity, psi,
 k = thermal conductivity, Btu/hr-ft- $^{\circ}$ F,
 ν = Poisson's ratio,
 q = heat generation per unit volume, Btu/hr-ft 3 ,
 b = outside radius of the tube, ft,
 a = inside radius of the tube, ft, and
 T = temperature at the surface denoted, $^{\circ}$ F.

Stresses due to transmitted heat will be expressed by a thermal gradient across the cylinder wall.

8.832 Stresses Due to Pressure

When a hydrostatic pressure is applied to either surface of the cylinder, the tangential stress is computed by the formula:

$$S = PR/h,$$

where

S = the tangential stress, psi,
 P = internal pressure, psi,
 R = inside radius, in., and
 h = thickness, in.

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1. R. Daane, "Nuclear Engineering Handbook", edited by H. Etherington, p. 9-112, 1958.

For an external pressure P is given a negative sign, but the stresses found are treated as significant only when the pressure is insufficient to cause failure through buckling (elastic instability).

The thickness of conventional vessel shells under internal and external pressure was obtained following the ASME Code. The formula for obtaining the thickness of vessel wall to withstand a given internal pressure is given as:

$$t = \frac{PR}{Se - 0.6P}$$

where

P = pressure (psi),
R = vessel radius (inches),
S = allowable stress in appropriate table, Section VIII, ASME Code (psi), and
e = joint efficiency in Table UW-12, Section VIII, ASME Code.

The Bureau of Ships exception of multiplying the factor B (ASME Code) by 1.33 for assigning wall thickness of members subjected to external pressure was used.

8.833 Combined Stresses

When the stresses in a structure occur as a result of more than one type of load, the resultant stress in the material is obtained by superposition of the primary or colinear stresses. The tangential pressure and thermal stresses can therefore be added algebraically to find the combined stress. If this stress is the numerical maximum, it can then be compared with the allowable stress values at temperature as required by the ASME, "Boiler and Pressure Vessel Code".

8.84 Reactor Core Vessel

In the ATR the diameter of the pressure vessel limits the use of heat treated high strength aluminum alloys, although these alloys have a distinct advantage in meeting ASME Code requirements at the higher internal heating rates. However, as the pressure vessel is moved away from the fuel annulus and the heating rate decreases, the advantage changes to the use of stainless steel (Type 304L or 347) because of its proven ability as construction material in reactor pressure vessels. The predictable better resistance to stress corrosion conditions under irradiation makes Type 304L or 347 stainless steel the preferred material at low (<0.4 watts/g) heating rates. Table 8.8C summarizes the vessel wall thicknesses and stresses for various internal heating rates at the following design conditions:

Internal pressure	300 psi
Metal temperature	200°F
Wall temperature difference	10°F
Welding efficiency (except as noted)	80 per cent

TABLE 8.8C
COMPARISON OF TOTAL STRESS WITH ALLOWABLE STRESS
FOR VARIOUS MATERIALS AND VESSEL DIAMETERS

Material	Internal Heating Watts/g	Inside Diameter Inches	Thickness* Inches	Total Stress psi	Allowable Stress** psi
Aluminum Alloys					
5154-0	10.0	39.0	1.1	11,500	11,000
5154-0***	10.0	39.0	0.9	11,000	11,000
5154-0	0.11	59.0	1.6	5,600	11,000
5154-0	0.02	72.0	1.9	5,700	11,000
Stainless Steel					
Type 347	10.0	39.0	0.4	43,600	28,100
	0.11	59.0	0.6	17,400	28,100
	0.02	72.0	0.8	15,600	28,100
Zircaloy-2****	10.0	39.0	0.5	22,300	22,500

* Thickness equal to or greater than ASME Code requires.

** Allowable stress is 1.5 times S value in Table 23, Section VIII, ASME Code.

*** Values on this line are based on a welding joint efficiency of 100 per cent.

**** This material is not listed by ASME Code, Section VIII (1959).

8.85 D₂O Reflector Tank

The D₂O reflector tank surrounds the reactor core (See Fig. 2.0A), so that the inner wall has a high internal heating rate (10 watts/g), and an unconventional pressure vessel shape. Certain design criteria were investigated to ascertain whether a conventional cylindrical pressure vessel could meet ASME Code requirements (Figures 8.8A and 8.8B). Aluminum alloy (5154-0) and zircaloy-2 were investigated. An allowable stress value for zircaloy-2 was obtained according to the basis of Appendix Q, Section VIII, ASME Code and data from Reactor Handbook of Materials, Second Edition as 15,000 psi at 300°F. The design conditions which will

allow a conventional vessel to meet requirements are:

Internal pressure	300 psig
External pressure	0 psig
Internal heat generation	10 watts/g
Inside diameter	39 inches

The corresponding wall thickness which the ASME Code requires for aluminum alloy 5154-0 for internal pressure only and welding joint efficiency of 100 per cent is 0.9 in., Table 8.8C. Although the wall thickness of zircaloy-2 for the design conditions previously given is 0.5 in., it is not listed for pressure vessel material by Section VIII, ASME Code, 1959.

In operation the D₂O tank walls may separate the process water system from the heavy water system. The possibility of loss of internal pressure on aluminum alloy tank walls for this case was investigated. The ASME Code requirements as modified by the Bureau of Ships exception for the thickness are given in Table 8.8D for the condition of loss of internal pressure and conventional cylindrical shape. The assumption of negligible internal heat generation and hence negligible thermal stress leads to conservative design because the heat capacity of the metal retains a temperature distribution in the metal so that a tensile stress in the outer wall remains which subtracts from the buckling stresses.

TABLE 8.8D

THICKNESSES OF D₂O

ALUMINUM ALLOY 5154-0 VESSEL WALLS

	External Pressure psi	Thickness (inches)
Inner wall (39.0 in. inside diameter)	150 300	1.1 2.0
Outer wall (72 in. inside diameter)	150 300	1.8 3.3

The inner tank wall with thickness desired by nuclear considerations (approximately 1.0 in.) can be designed to conform to the four-lobe shape (Fig. 2.0A) only by alteration of the above design condition of possible loss of internal pressure and provision of suitable rigid members to fix the four-lobe shape and reduce the wall stresses.

Since both sides of the wall are in contact with water (heavy or light) the internal and external pressure may be made to correspond closely, or the external wall pressure from D₂O could be maintained in an intermediate range (100 - 150 psi).

8.86 Beryllium Reflector of Alternate Arrangement

For the alternate arrangement, Fig. 2.0B, with a beryllium reflector, an acceptable service lifetime of the beryllium before swelling or crumbling is the primary consideration as the pressure vessel diameter can be great enough to place it in a region of low internal heat generation. The major variables affecting this service life are the neutron exposure, the rate of exposure, and the temperature of the beryllium. Previous experience with MIR beryllium in the M-6 reflector position indicates satisfactory service to the present time for a calculated neutron exposure of about 2×10^{22} nvt (> 1 mev). The calculated maximum exposure to which the beryllium in the vicinity of the four-lobe core will be exposed is $(1.5 - 2.0) \times 10^{22}$ nvt (fast) per year, hence after allowance for the rate effect and difference in temperature one would predict that the service life of the beryllium nearest to the ATR core would be approximately one year. Annealing experiments¹ on MIR shim rod beryllium also lead to a prediction of service life in excess of one year with swelling less than 1 per cent.

In the ATR alternate arrangement metals such as aluminum or zirconium can replace the beryllium in the highest flux positions which are in the cusps of the four-lobe core so that the highest exposure the beryllium will experience will be less than the $(1.5 - 2.0) \times 10^{22}$ nvt per year. The irradiation exposure of the beryllium also decreases with distance so that at 1.6 cm into the beryllium the annual exposure has dropped to approximately 1.0×10^{22} nvt (fast). Thus the major amount of beryllium would have a service lifetime in excess of two years.

8.87 In-Pile Tubes

In order that the entire loop system containing primary coolant may be decontaminated by various chemical procedures, the in-pile loop tubes should be constructed of high alloy steels.

The pressure and thermal stresses for loop tubes of various thicknesses are given for the following design conditions:

Internal pressure	2200 psig
External pressure	0 psig
Uniform internal heat generation	20 watts/g
Maximum metal temperature	500°F
Internal diameter	3.018 in.

The requirement of maximum loop water coolant temperature of 650°F can be met by providing an adiabatic inside wall (stagnant water layer or gas barrier) so that the thermal stresses were calculated for

1. C. E. Ells, "The Swelling of Beryllium From Neutron Induced Gases", C R Met-809, November, 1958.

cooling from the outside wall only. A comparison of total stress with allowable stress¹ (1.5S) is given in Table 8.8E for stainless steel.

TABLE 8.8E

COMPARISON OF TOTAL STRESS WITH ALLOWABLE STRESS

Material	Inside Diameter in.	Thickness* in.	Total Stress psi	Allowable Stress** psi
Type 347 ss	3.018	0.241	82,800	22,800
Type 316 ss	3.068	0.216	70,000	25,800

* Thickness equal to or greater than required by ASME Code.

** Allowable combined stress is 1.5 times S value from Table 23, Section VIII, ASME Code.

It is seen from the above table that with these design conditions even the strongest (Type 316) 3 in. stainless steel schedule forty pipe (0.216 in. wall thickness) does not meet the allowable stress requirements of the ASME Code; neither will a thicker wall. Although construction of the loop tube from Type 347 stainless steel requires a thicker wall to meet the pressure stress requirements, its known corrosion and fatigue life behavior under irradiation makes it the preferred material of construction for the loop tubes. Reduction of the thermal stress even by reduction of the temperature difference across the wall to 10°F, does not result in a combined stress within ASME Code. Design of the loop tubes with high thermal heating rates must therefore be made in accordance with tentative methods involving allowable-pressure and tentative-fatigue design procedures.

Stainless steel loop tubes with design characteristics similar to those indicated in Table 8.8E are now being used successfully in reactors. The justification entails calculation of the fatigue life from the strain cycling that occurs since the combined stresses exceed the proportional limit, e.g., the yield strength for Type 316 stainless steel at 500°F is 27,000 psi.² This design procedure has not as yet been accepted by

1. Interpretations of ASME Boiler and Pressure Vessel Code, Case 1273N.
2. Reactor Handbook of Materials, U.S. AEC, McGraw-Hill Printing Company p. 571, 1960.

ASME but is used by the Bureau of Ships.¹ Following the procedure of Coffin,² the fatigue life for Type 347 stainless steel under the conditions of Table 8.8E is 65,000 cycles. For a yield stress of Type 347 stainless steel of 32,000 psi and pressure and thermal stress values for a thickness of 0.241 in. the progressive expansion of the loop tubes may be examined following the method of Miller³ with the result that cyclic growth does not occur for a linear or parabolic temperature variation in the metal.

From the above considerations the loop tube should be constructed of Type 347 stainless steel with a thickness of 0.241 in. for the 3 in. i.d. tubes.

-
1. "Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components", dated December 1, 1958, Bureau of Ships.
 2. L. F. Coffin, "Design Aspects of High Temperature Fatigue with Particular Reference to Thermal Stresses", Trans. ASME Vol. 78, p. 527, 1956.
 3. D. R. Miller, "Thermal-Stress Ratchet Mechanism in Pressure Vessels", KAPL-1955, August 12, 1958.

FIG. 8.8A
 STRESSES IN CONCENTRIC D₂O VESSEL

FIG. 8.8B
 TANGENTIAL STRESSES IN REACTOR PRESSURE VESSEL
 WALL THICKNESS

PP CO.-C-3042