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REACTORS—RESEARCH AND TESTING

UNITED STATES ATOMIC ENERGY COMMISSION

PRELIMINARY INVESTIGATIONS FOR AN
ADVANCED ENGINEERING TEST REACTOR

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February 22, 1957

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PRELIMINARY INVESTIGATIONS

FOR AN

ADVANCED ENGINEERING TEST REACTOR

Work Performed for the
Chicago Operations Office
U. S. Atomic Energy Commission

Contract No. AT(11-1)-500

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Abstract

A study was carried out to determine a suitable reactor system or systems for the testing of water, gas, and liquid-metal cooled fuel assemblies. The unperturbed thermal neutron fluxes in the test holes are to be $1 - 1.5 \times 10^{15}$ n/cm²sec. A group of seven separately cooled and controlled reactors assembled in a common shield are proposed to meet the test facility requirements. Each of the reactors consists of the central test thimble, surrounded consecutively by an H₂O annulus, reactor core, and D₂O reflector. The estimated total power of the seven reactors is about 450 MW when the neutron flux requirements are met in all the test holes.

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1.0 INTRODUCTION

In December, 1956, the Chicago Operations Office of the Atomic Energy Commission requested Internuclear Company to conduct a survey of reactor concepts and to prepare a report with recommendations for a reactor system or systems which would meet certain requirements for an Advanced Engineering Test Reactor. The requirements, which are given in more detail in Appendix 1.0, consist primarily of unperturbed thermal neutron fluxes of $1 - 1.5 \times 10^{15}$ n/cm²sec in water, gas, and liquid metal cooled loops and a perturbed thermal neutron flux of $4 - 5 \times 10^{14}$ n/cm²sec in the water cooled loop.

After preliminary discussions with reactor engineers and physicists of Argonne National Laboratory, Oak Ridge National Laboratory, and other organizations prominent in the reactor field, it was decided to concentrate the efforts of this study on H₂O and D₂O cooled and moderated heterogeneous reactors. The basic technology for such reactors is well proven and thus a reliable operational reactor is assured if a suitable reactor arrangement can be developed. The choice is further substantiated by the fact that a reactor for component testing should not in itself be a developmental experiment.

2.0 RECOMMENDED REACTOR SYSTEMS

The primary purpose of the Advanced Engineering Test Reactor (AETR) is the testing of fuel assemblies in high thermal neutron fields. In order to achieve significant advances over test loop facilities in present test reactors, the test loops in the AETR were selected as the focal points in design and various arrangements of individual reactor cores around them were investigated. The goal in the studies of the individual reactor cores was the achieve the desired neutron fluxes in the test loops with minimum reactor power. Although not a requirement for the AETR, the studies showed that unperturbed thermal neutron fluxes of 5×10^{15} n/cm²sec could be achieved with reactors of about 125 MW power.

2.1 General Recommendations

Below are given general recommendations for an Advanced Engineering Test Facility.

2.1.1 Overall Reactor Facility

It is recommended that the AETR consist of seven separate reactor systems, one for each major fuel element test loop. The reactors associated with the systems would be assembled in a common shield but would be shielded from each other in an arrangement similar to that shown by Figure 2.A. Each of the reactors would be entirely independent of each other with respect to control and cooling, except that it appears feasible to use a common secondary cooling system.

Desirable features of an AETR consisting of separate reactors for each loop are as follows.

1. Fuel assembly testing can be carried out under optimum conditions since each test is entirely independent of the others. The reactor can be started or shut down and reactor power level can be varied to suit the individual experiment without any concern for effects on other tests. Since the reactors are all shielded from one another, it is possible to carry tests to destruction of the test element without seriously endangering the entire reactor facility. Neutron flux distributions in the test elements are not subjected to perturbations caused by changes in other test holes.

2. In the seven-reactor system only those reactors containing experiments are normally operated, while a single large reactor with multiple in-core test holes wastes fuel if any of the test holes are not occupied by experiments. Considering that the cost of U^{235} burnup for a 500 MW fully enriched reactor is about \$9,000/day, a sizable saving in fuel costs may be realized.
3. The reactor power requirements for the seven separate reactors are comparable to, or even less than that for a single reactor with comparable test holes. For example, the high flux test space in the seven separate reactors proposed herein is comparable to that in the five in-core test holes in the ETR, yet scaling the ETR power up to provide comparable flux would result in a reactor power of about 850 MW compared to about 480 MW for the seven-reactor system.
4. In the reflector-moderated reactors of the seven-reactor system, fuel burnup has less effect on reactivity than in-core moderated fully enriched reactors and thus relatively long reactor core lives and high percentage fuel burnups are achievable.
5. The AETR design concept outlined in this report allows for easier access for changes in experiments and for changes in fuel elements. The test holes, cores, and control system are geometrically separated, expediting operation and maintenance.

An alternate to use of separately shielded reactors would be to assemble the individual reactor cores in a single large D_2O or H_2O filled tank. However, it is anticipated that the design problems regarding the different loop pressures and associated stresses (thermal and pressure) would be much more difficult to solve, and operation of the system would be more inefficient and complicated.

2.1.2 Reactor Unit

It is recommended that each of the individual reactor units consist primarily of the test element surrounded consecutively by annular regions of H_2O , fully enriched reactor core, and D_2O reflector separated by appropriate pressure shells. Typical reactor arrangements for water cooled and gas cooled loops are

shown by Figures 2.B and 2.C, respectively. These are discussed in more detail in Section 2.2. The annular reactor core should be three feet long and of minimum size consistent with ability to achieve the necessary fuel loading and heat removal. The reflector thickness should be kept to a minimum consistent with ability to achieve proper reactor control.

The fuel assemblies should be aluminum plate-type fuel assemblies with U-Al alloy or UO_2 -Al meat. Figure 2.E shows a flat-plate fuel assembly which could be built with present technology. Figure 2.D shows a plate-type fuel assembly which would require development, but offers an advantage over the flat-plate (Figure 2.E) assembly in that the annular core region is completely filled with fuel assembly. However, the flat plate assembly (see Figure 2.E), which has the plates oriented approximately radially, has an advantage over assemblies with plates oriented circumferentially (see Figure 2.D) in that lateral transfer along the plate will help to reduce the maximum surface heat flux. As will be seen later the neutron flux rises sharply at the inner diameter of the reactor core - the heat flux at the inner core diameter is generally 1.3 to 1.5 times what it is one centimeter into the core. With a fuel assembly similar to that in Figure 2.D, lateral transfer of heat to the side plates and along the plate will reduce the maximum heat flux appreciably, while mixing in the water channel will reduce the bulk water temperature at the high heat flux region at the inner diameter. Special efforts should be made to develop a fuel assembly that can be fabricated cheaply since large numbers of them will undoubtedly be required. Unless it is not practical with respect to required reactor power or other factors, all reactors should be designed to use the same size fuel assembly. Use of aluminum rather than stainless steel fuel assemblies is recommended primarily on the basis of the relative costs of these assemblies. There appears to be little reason to operate the fuel assemblies at temperatures above those permissible with standard aluminum alloys or the aluminum-nickel alloy ($\sim 1\%$ Ni) recently developed by Argonne National Laboratory. The latter can be operated at temperatures in the 500 to 550°F range while standard alloys are satisfactory for temperatures as high as 425°F for the relatively short (1 to 4 weeks) reactor core lives that are contemplated for full power operation of the AETR reactor cores.

2.1.3 Reactor Pressure and Temperature

In general an operating pressure of about 300 psi for the primary loops of the reactors is recommended. This would permit use of aluminum fuel assemblies of standard alloys (2S for example) if the maximum fuel plate temperature is limited to the saturation temperature at operating pressure (417°F at 300 psi). Also, since five of the test loops are to operate at 300 psi, this minimizes the pressure difference between the reactor and test loops during normal operation - this is an important consideration with regard to pressure and thermal stresses in the pressure shells. Because of the latter it may be necessary to operate the reactors for the 2000 psi water-cooled test loops at higher than 300 psi.

The normal operating temperature of the reactor coolant should be as low as is consistent with final disposal of the reactor heat via cooling towers, a spray pond, or other means. The low bulk water temperatures are desirable since they permit higher reactor power densities, and also because the burnout heat flux increases appreciably with sub-cooling ($t_{\text{sat}} - t_{\text{bulk}}$). A bulk water temperature of less than 130°F at the reactor inlet appears both desirable and feasible.

2.1.4 Reactor Coolant

Use of both D₂O and H₂O as reactor core coolants have been investigated. The reactor power required to meet neutron flux specifications in the test hole is about 20 to 30% less with D₂O cooled reactors than with H₂O cooled reactors of identical geometry. However, the H₂O cooled reactors have about 8 to 15% more reactivity than the D₂O cooled reactors. Since both the required reactor power and the reactivity decrease with decreasing reflector (D₂O) thickness, an H₂O cooled reactor can have a thinner reflector (D₂O) than an identical D₂O cooled reactor of the same reactivity. Thus it appears that the required reactor power with an H₂O cooled core may be only about 10 to 20% higher than that with a D₂O cooled core of identical reactivity. While the higher reactor power required with H₂O cooled cores leads to higher fuel costs, the high fuel costs are somewhat offset by savings in D₂O inventory and design and construction costs. Also of importance is the additional complications encountered in unloading

and loading the core of the D₂O cooled reactor. The relative merits of H₂O versus D₂O as the reactor core coolant needs careful investigation before the best choice can be made.

2.1.5 Reflector

D₂O is recommended for the reflector. Since the reactor cores are small and are primarily reflector-moderated, use of an H₂O rather than a D₂O reflector requires that the reactor core size or loading be increased appreciably to compensate for reactivity losses. Also, the H₂O reflector is not as satisfactory for reflector control since the reflector gives less reactivity change. The D₂O reflector provides a larger volume, while the H₂O reflector provides a higher neutron flux for capsule irradiations in the reflector.

Solid moderators were not investigated because of the desire to use reflector control during reactor operation.

2.1.6 Reactor Control

In a reactor for fuel assembly testing it is particularly desirable to minimize neutron flux perturbations, due to reactor control, in the test hole. With this in mind it is recommended that the reactor be reflector-controlled. Two schemes for reflector control are shown by Figures 2.F and 2.G.

Figure 2.F shows a system wherein the shim control is provided by borating the outer portion of the reflector while reactor "scram" is accomplished by dropping the "pure" D₂O in the inner annulus. The borated D₂O is continuously circulated through a heat exchanger with a by-pass stream passing through an ion-exchanger for removal of boron to compensate for fuel burnup. A boric acid injection system in the main stream provides for additions of poison to reduce reactivity gradually. Initially, before reactor startup, the outer annulus is heavily borated and the inner "pure" D₂O annulus is down. The inner annulus is raised by starting the pump, which circulates the reservoir D₂O through a water-jet eductor, which draws a vacuum on the top of the inner annulus and raises the water level therein. When the water reaches the

top of the annulus, water is drawn through the heat exchanger, the eductor, and returned to the reservoir. This provides continuous circulation of the D_2O during reactor operation to remove heat generated in the D_2O by gammas and slowing down of neutrons. Reactor "scram" is achieved by opening the scram by-pass valve, which by-passes the eductor and permits the D_2O level in the inner annulus to drop. The number of safeties is increased by increasing the number of by-pass valves in parallel. Loss of pump power also results in reactor shutdown.

Figure 2.G shows a reflector control scheme in which the reflector is raised in successive annuli proceeding outward from the reactor core. Normally the reflector is fully raised in the inner annulus and partially raised in the next annulus before the reactor becomes critical. Fuel burnup is compensated for by increasing the level in each succeeding annulus until all annuli are completely full. The systems for each annuli are similar to those described above for the inner annulus of the control system shown by Figure 2.F except that control valves are incorporated in the systems for the outer annuli. These valves control the water level in the annuli by controlling the suction head of the eductors. Until the water levels in the outer annuli reach the top, internal heating may cause boiling of the D_2O in these regions, in which event the effluent steam is condensed in the heat exchangers and is returned to the reservoirs.

2.2 General Characteristics of Recommended Systems

Although the optimum reactor core diameters are smaller for the small water-cooled test holes than for the large gas-cooled test loops, in the interest of standardization all reactor cores should be of the same size. Based on the above philosophy, Table 2.2 gives a summary of estimated reactor powers necessary to satisfy the requirements given in Appendix 1.0. The reactor powers required to meet specifications are given for both H_2O - cooled and D_2O - cooled reactors. As previously emphasized - the choice of coolant should await further investigation. It should be noted that the power requirements for the gas and liquid metal cooled loop reactors are based on the unperturbed neutron flux in an empty test hole. If the unperturbed flux was based on a water filled hole, the power requirements would be substantially less, probably about half the values given.

Table 2.2 Estimated Reactor Power for Reactors

<u>Requirements</u>	<u>Water-Cooled Test Loops</u>		<u>Gas and Liquid Metal-Cooled Test Loops</u>		
Test hole I.D., in.	3	4	3	4.5	6.8
Number required	1	1	1	1	3
Unperturbed ϕ_{TH} Av. re- quired, n/cm ² sec	10 ¹⁵	10 ¹⁵	10 ¹⁵	1.5 x10 ¹⁵	1.5 x10 ¹⁵
Temperature, °F	600	600	1500	2200	2200
Pressure, psi	2000	2000	300	300	300
<u>Estimated Design Data</u>					
Reactor core I.D., in.	12	12	12	12	12
Reactor core O.D., in.	16	16	16	16	16
Reactor core height, in.	36	36	36	36	36
Reactor power to meet neutron flux specifications in					
D ₂ O cooled reactor core, MW	30	30	40	80	100
H ₂ O cooled reactor core, MW	35	35	48	96	120

From the above table, the estimated total reactor powers for the seven reactors are about 480 and 570 MW for the D₂O cooled and H₂O cooled cores, respectively. By optimizing the size of each reactor with respect to its test hole, the above could probably be reduced about 10%. However, it appears more logical to design all reactors with the same size test hole and for the same maximum power. It would then be possible to achieve unperturbed thermal neutron fluxes of 3 to 4 x 10¹⁵ n/cm²sec in water cooled loops. The total U²³⁵ loading of the seven reactors would be about 35 to 45 kg which, although high, is not particularly undesirable since inventory charges are not a significant part of operating costs.

2.2.1 Typical Gas-Cooled Loop Test Reactor

A typical gas-cooled loop test reactor is shown by Figure 2.C. The center of the reactor is occupied by a gas-cooled test thimble, which is inserted from the bottom of the reactor. The test thimble is separated

from the reactor water by an air gap and an aluminum thimble which forms the inner reactor wall. The core cooling water flows up the water annulus inside the core and then down thru the reactor core. An aluminum pressure vessel separates the reactor core from the D₂O reflector. The arrangement shown is for an H₂O cooled reactor core. If the core is D₂O cooled, it would be necessary to bring the core coolant in at the top and provide a separate H₂O loop for the H₂O annulus between the reactor core and test hole.

Aluminum pressure vessels are used primarily because of their good thermal conductivity and low thermal neutron absorption. Both these properties enhance the suitability of aluminum for the reactor pressure walls between the reflector and core and between the test thimble and core. Since both these vessels are relatively small (approximately 7 inch and 15 inch diameter, 6 feet long), it is feasible to fabricate them from aluminum forgings. With 130°F or lower temperature inlet cooling water, the aluminum wall temperatures could be expected to remain below 250°F. For this application it appears feasible to use CS41A-T6 aluminum, which has a tensile strength of 65,000 psi, a yield strength of 55,000 psi, and an allowable stress of 14,000 psi at 250°F. While the thermal and pressure stresses in the reactor vessel walls were not analyzed thoroughly, preliminary calculations indicate that use of aluminum forgings for these components is practical. Use of an aluminum pressure wall between the core and reflector permits the reflector to be operated near atmospheric pressure.

The arrangement of the reactor thimble and the test thimble is such that the space between them is operated at the same pressure as the test loop. This permits use of a thin-wall for the test thimble since there is essentially no pressure differential across it. Although this arrangement may not be necessary for the 300 psi loops, it would be necessary for the high pressure (2000 psi) test loops, since the high thermal stresses (from gamma heating) necessitate use of relatively thin walls.

Nuclear Considerations

The reactor arrangement described above was selected on the basis of results from nuclear

studies of various arrangements. The results of the studies are given by Problems 33.02a through 37.52a (see Appendix 2.0). The reactor arrangement shown by Figure 2.C is identical to that of Problem 35.02a except that the reflector thickness is 30 cm. rather than 20 cm. The radial neutron flux distributions for Problem 35.02a are shown by Figure 2.H. The thermal neutron flux peaking will be slightly less for the 30 cm reflected reactor than that in Figure 2.H. Comparisons of the thermal neutron flux distribution in a similar reactor with and without a test section are shown by Figure 2.I.

Estimated reactor data for the reactor shown by Figure 2.C are given in the following table, where the reflector thickness of the H₂O cooled reactor has been reduced to give a k comparable to that for the D₂O cooled reactor.

Table 2.2.1 Estimated Reactor Data for Gas-Cooled Loop Test
Reactor

	<u>D₂O Cooled</u> <u>Core</u>	<u>H₂O Cooled</u> <u>Core</u>
Reactor Core:		
Volume, liters	39.1	39.1
Kg U ²³⁵	6.13	6.13
M/W ratio	1.0	1.0
N _U - atoms U ²³⁵ /cc	4 x 10 ²⁰	4 x 10 ²⁰
Reflector (D ₂ O) thickness, cm	30	23
Reactivity, k	1.04	1.04
Radial max/av power ratio	1.37	1.26
Reactor power for 1.5x10 ¹⁵ ϕ _{TH} av in test region, MW	97	115
Maximum core power density for 1.5x10 ¹⁵ ϕ _{TH} av in test region, MW/liter	4.1	4.5

The data reported for problems 34.02a, 35.02a, and 36.02a show that the least reactor power is required with an H_2O annulus about 6 cm thick between the empty test hole and the reactor core.

Heat Transfer

Table 2.2 gives maximum core power densities of 4.1 and 4.8 MW/liter for the D_2O cooled and H_2O cooled cores, respectively. These are appreciably higher than the 2.75 MW/liter permissible power densities given by Figure 4.A for 30 ft/sec D_2O velocity in 60 mil channels between 50 mil plates, at 300 psi. However, by taking advantage of lateral transfer to the side plates in the fuel assembly shown by Figure 2.E, by increasing the water velocity to 35 to 40 ft/sec, and by relaxing somewhat on the hot channel factors (see Appendix 4.0), a maximum power density of 4-5 MW/liter may be achieved. Relaxation of the hot channel factors used in the calculations seems justified on the basis of the entirely satisfactory operating experience of the MTR with essentially no hot channel allowance for fabrication uncertainties, and since local boiling over a small area should not cause channel plugging in the fuel assembly shown by Figure 2.E.

2.2.2 Typical Water Cooled Loop Test Reactor

The typical water cooled loop test reactor described herein is for a smaller reactor core than that recommended in Table 2.2 and thus represents a more optimum arrangement for meeting the specifications. The general arrangement, except for component diameters, would remain the same for the larger reactor. The radial arrangement is the same as that for Problems 29.04a, 29.04b, and 29.54b (see Appendix 2.0).

The general arrangement for the reactor, which is shown by Figure 2.B, is similar to that of the gas-cooled loop test reactor (Figure 2.C) except that the H_2O annulus between the test section and core is incorporated in the test thimble instead of in the reactor vessel, and the inlet water enters at the reactor top. The reactor vessel walls are aluminum forgings. The reactor thimble is designed to withstand 2000 psi loop pressure, thus permitting use of

thin-walled stainless steel for the test thimble - the air gap between these two thimbles is tied to the pressurizer of the test loop so that the pressure in it is always near the test loop pressure. This arrangement, which uses an aluminum forging for the pressure wall separated from a thin-walled stainless steel test thimble by an air gap thermal barrier, lessens the problems associated with operation of high pressure water-cooled loops in a reactor. The outer wall of the reactor vessel as shown by Figure 2.B is substantially thicker than it need be - this thickness conforms to that shown on the neutron flux distributions.

Nuclear Considerations

The reactor arrangement described above was based on nuclear studies of various arrangements, the results of which are given by Problems 29.04a thru 32.52b in Appendix 2.0. Figure 2.J and Figure

2.K show radial distributions of the fast, intermediate, and thermal neutron fluxes in the D₂ cooled typical reactor without and with test section, respectively. Figure 2.L shows thermal neutron flux distributions for both D₂O cooled and H₂O cooled reactors.

Reference to Problems 29.04b and 29.52b shows that the H₂O cooled reactor requires about 30% more power than the D₂O cooled core but has about 10% more reactivity.

Estimated reactor data for the reactor shown by Figure 2.B are given in the following table, where the reflector thickness of the H₂O cooled reactor has been reduced to give a k comparable to that for the D₂O cooled reactor.

Heat Transfer

Since the maximum power density in the core is only about 1 MW/liter, the heat can be readily removed with 30 ft/sec D₂O at 300 psi (see Figure 4.A). Since maximum power densities of about 4 to 5 MW/liter should be permissible with water at 35 - 40 ft/sec the reactor described herein could be operated at about 100 MW and would give

Table 2.2.2 Estimated Reactor Data for Water-Cooled Loop
Test Reactor

	<u>D₂O Cooled</u> <u>Core</u>	<u>H₂O Cooled</u> <u>Core</u>
Reactor Core:		
Volume, liters	29.9	29.9
Kg U ²³⁵	4.7	4.7
M/W ratio	1.0	1.0
N _U - atoms U ²³⁵ /cc	4 x 10 ²⁰	4 x 10 ²⁰
Reflector (D ₂ O) thickness, cm	30	
Reactivity, k		
With T-1 test section	1.11	1.11
Without test section	0.96	-
Radial max/av power ratio	1.34	1.29
Reactor power for 5x10 ¹⁴ ϕ_{TH} av in test section, MW	25	30
Reactor power for 10 ¹⁵ ϕ_{TH} av in test region (no test section), MW	22	26
Maximum core power density for 5x10 ¹⁴ ϕ_{TH} av in test section, MW/liter	0.94	1.1

unperturbed thermal neutron fluxes of about
 4×10^{15} n/cm²sec in the test hole.

While the above reactor has been designed to achieve highest thermal neutron flux with minimum reactor power, by sacrificing slightly on reactor power the core could be increased in size to reduce the reactor power density per MW reactor power (see Section 3.2.1). It is estimated that

by doubling the reactor annulus thickness and reducing fuel concentration accordingly, it is feasible to design a reactor that would operate at about 250 MW and give a thermal neutron flux near 10^{16} n/cm²sec in the test hole.

2.2.3 Liquid Metal-Cooled Loop Test Reactors

Reactor calculations involving liquid metal coolants in the test hole were not made. However, the reactor arrangement for the liquid-metal cooled test loops would be similar to that for the gas-cooled test loops. Also, it is anticipated that reactor power requirements for similar test volumes and neutron fluxes would be about the same as for the gas-cooled test loops.

2.2.4 Space for Capsule Irradiations

The seven reactors offer a relatively large volume of high neutron flux for capsule irradiations since both the H₂O annulus and D₂O reflector can be used for this purpose. As can be seen by reference to Figure 2.H, when there is no test section in the test hole the maximum thermal neutron flux in the H₂O annulus is about 3.8 times what it is in the D₂O reflector. However, with a test section in the test hole this ratio would drop to about three. Small capsules (say < 1" diameter) could be incorporated in the water annulus between the core and test hole without seriously perturbing the test hole neutron flux. These would have neutron flux fields of about 1.5×10^{15} n/cm²sec at full reactor power (~100 MW). Larger capsules would be placed in the D₂O reflector, where the neutron flux would be about 5×10^{14} n/cm²sec.

2.2.5 Space for Irradiation of Large Components

Space for irradiation of large components (say 3-foot cube) would be provided between some of the reactors, as indicated by Figure 2.A. If these spaces are incorporated in the AETR, the reactor control scheme involving borating of the D₂O would not be practical, and the level control method would be recommended. The neutron levels in such spaces are estimated to be about 10^{12} to 10^{13} n/cm²sec.

3.0 REACTOR SURVEY SUMMARY

A general survey of reactor types suitable for meeting the requirements given in Appendix 1.0 was undertaken. Because of the relatively advanced stage of technology enjoyed by water-cooled heterogeneous reactors over other types and also because of the relatively high power densities attainable in such reactors, the study was limited to D_2O and H_2O cooled heterogeneous reactors.

The problem was approached from the standpoint of designing small individual reactor cores around the test loops, rather than attempting to fit test loops into a single large reactor core. This approach is valid irrespective of whether the small reactor cores are assembled into a single large reflector-moderator to form a single reactor or constructed as entirely separate reactors. Since, assuming the same fuel density, both H_2O cooled and D_2O cooled reactors can be made smaller with D_2O reflector than with H_2O reflector, the studies were limited to D_2O reflected cores. The various reactor cores were compared on the basis of meeting the neutron flux requirements in the test loop with minimum reactor power.

It was not possible, within the time available, to compare all reactors at the same multiplication constant. However, since the comparisons are based on neutron flux distributions, it is believed that they are, in general, valid irrespective of the differences in reactivity. The values of the neutron multiplication constant for all problems investigated are given in the tabulations in Appendix 2.0.

Well over one hundred reactor cores involving different sizes, arrangements, and compositions were analyzed. The reactor calculations were carried out with an IBM-650 computer.

3.1 Reactor Cores with D_2O Islands

Extensive investigations of reactor cores with D_2O islands were unnecessary because of recently reported work on a high flux research reactor⁽¹⁾. Also, preliminary investigations of reactor cores with light water islands showed that they are undoubtedly superior for the present application. Several reactors with D_2O islands were analyzed

(1) ORNL-CF-56-8-206, "High Flux Research Reactor", R. D. Cheverton, et al

primarily for direct comparison with reactors with the H_2O islands. These are problems 1.01a through 3.01b in Appendix 2.0. Since D_2O islands have to be quite large to attain appreciable peaking of thermal neutron flux in the center of the island, the reactors investigated had islands of 24 cm and 40 cm radius. Typical radial neutron flux distributions for several of the D_2O island reactors are shown by Figures 3.A, 3.B, and 3.C. As will be noted, all neutron flux distributions in this report are plotted in n/cm^2sec per megawatt of reactor power. This permits direct comparison of reactors on the basis of which reactors provide the highest thermal neutron fluxes in the test loops for the least reactor power. Comparison of Figures 3.A and 3.C and also the data for these reactors given in Appendix 2.0 shows that as the island dimensions decrease, the maximum thermal neutron flux in the island increases. Comparison of Figures 3.A and 3.B shows a similar effect for decreasing core thickness and associated increase in core fuel concentration. However, as will be shown later, the D_2O islands are not as good as the H_2O islands in the smaller reactor cores.

The D_2O island has an advantage over H_2O islands where a relatively large volume at high thermal flux is desired for the irradiation of small capsules. The thermal neutron flux in the D_2O island can be improved appreciably by use of a light water annulus between the D_2O island and the reactor core. This can be seen by comparison of Figures 3.B and 3.D, the latter figure being for a reactor core with a 4 cm H_2O annulus between the reactor core and D_2O island. The optimum thickness for the H_2O annulus was not determined but probably lies between 2 cm and 8 cm. As can be seen by comparison of problems 18.01a and 27.01a, in small reactors a full H_2O island is better than a D_2O island followed by an H_2O annulus.

3.2 Reactor Cores with H_2O Islands

A parametric study of reactors with H_2O islands was undertaken wherein the size of the island, core size, fuel loading, and metal-to-water ratio in the core were varied. The large majority of the reactors were analyzed both with and without a water-cooled test assembly in the center.

3.2.1 Effects of Reactor Size and Fuel Loading

Reactors with fuel loadings of 0.5, 1.0, 2.0, and 4.0×10^{20} atoms U^{235}/cc were investigated. The thickness of the annular cores were 16, 10, 6, and 3.5 cm, respectively, for the above fuel loadings. The thick-

ness of the water annuli around the test section was varied from 2 cm to 8 cm. The results of the analyses are given by problems 4.01a, b, through 18.01a, b. All the cores in this phase of the parameter study were D₂O cooled with a metal-to-water ratio of 0.7.

Figure 3.E shows curves for reactor power required for an average unperturbed thermal neutron flux of 10^{15} in the test region ($r = 4$ cm) versus H₂O annulus thickness. Figure 3.F shows similar curves for a perturbed thermal neutron flux of 5×10^{14} in a typical water cooled test section and Figure 3.G shows similar curves for a power of 5 MW in the test section. All curves show a minimum power for a water annulus of about 4 cm. As can be seen from the curves, a reactor of about 20-25 megawatts would meet the requirements for the water-cooled test loop. While the power to produce 5 MW in the test section is relatively high, this is primarily a function of test assembly size, which could be increased in order to achieve a 5 MW output.

Figure 3.H shows curves for maximum power density in the reactor core necessary to achieve a neutron flux in the test section of 5×10^{14} n/cm²sec. Although the power densities are fairly high for the highly loaded small cores, they are quite readily achieved with water velocities of 30 ft/sec through flat plate assemblies operating at 300 psi pressure (see Chapter 4.0).

Figures 3.I, 3.J, 3.K and 3.L show radial thermal neutron flux distributions for the 3.5 cm thick core ($N_U = 4 \times 10^{20}$ atoms U²³⁵/cc) for 2, 4, 6, and 8 cm. thick annuli around the test hole. Although the unperturbed flux peak in the test hole is highest with the 2 cm water annulus, it drops off sharply and thus the average unperturbed thermal neutron flux in the test region is higher for the 4 cm water annulus.

Figures 3.J, 3.M, 3.N, and 3.O show radial thermal neutron flux distributions for reactor cores of thickness 3.5, 6, 10, and 16 cm, respectively, for reactors with a 4 cm H₂O annulus around the test hole. Comparison of these curves shows the relative advantages of the smaller highly loaded cores over the larger lightly loaded reactor cores.

3.2.2 Effects of D₂O and Void Between Test Hole and Reactor Core

The effects of a void and D₂O instead of H₂O between

the test hole and reactor core were investigated (see problems 19.01a through 26.01b). In all cases the test section was immediately surrounded by 2 cm H₂O which represented the incoming water in the test thimble, i.e., the test thimble water flows up around the outside of the test assembly and then down through the test assembly. The results are shown by Figure 3.P. As expected, the H₂O is superior to either D₂O or a void. However, judging from the slopes of the curves, there may not be very much difference between D₂O and H₂O in the annulus if the test assembly were surrounded by 4 cm of H₂O instead of only 2 cm.

3.2.3 Effects of Metal-to-Water Ratio in the Reactor Core

The effects of varying metal-to-water ratios in the reactor core were investigated for both D₂O and H₂O core coolants. The data are summarized under problems 5, 8, 11, 12, 13, 14, and 16.

Figure 3.Q shows curves for reactor power required for an unperturbed thermal neutron flux of 10^{15} n/cm²sec in the test hole versus M/W ratios in D₂O cooled cores. Figure 3.R shows similar curves for a perturbed thermal neutron flux of 5×10^{14} n/cm²sec in the test section. These figures show that M/W ratio is unimportant for the small highly loaded cores, and that the required reactor power decreases more rapidly with increasing M/W ratio in the H₂O cooled cores than in the D₂O cooled cores. Figure 3.R also shows that D₂O cooled cores require less reactor power to meet the neutron flux requirements than do H₂O cooled cores and that the difference in required power increases with reactor core size.

Figure 3.S shows curves for maximum power density in the reactor core versus M/W ratio for a highly loaded ($N_U = 4 \times 10^{20}$ atoms U²³⁵/CC) core. The maximum power densities required with H₂O coolant are about 20% higher than those required with D₂O coolant.

Figure 3.T shows curves for the neutron multiplication constant versus M/W ratio. The reactivity decrease with increasing M/W ratio is more pronounced with the H₂O cooled cores than with the D₂O cooled cores. However, the H₂O cooled cores have significantly higher reactivity than the D₂O cooled cores.

3.2.4 Effect of Reflector Thickness

In the course of the investigations for the typical reactors for gas-cooled and water-cooled loops, some comparisons were made of reactors with different reflector thicknesses. The results of the studies are given by Problems 29.52b - 32.52b, 33.02a, and 34.02a. The effect of reflector thickness on required reactor power of the water-cooled loop test reactor is shown by Figure 3.U. There is a gradual increase in the required reactor power with increase in reflector thickness. The required power levels off at a reflector thickness of about 60 cm, where it is about 25% higher than what it is with 18 cm reflector. Figure 3.V shows the effect of reflector thickness on reactivity for the same reactor.

4.0 HEAT TRANSFER

In order to obtain the desired neutron flux in the test hole of the reactor, the core will need to be operated at very high power densities. Since it was realized in the early stages of the reactor studies that the required neutron fluxes in the test holes could be attained more readily with D₂O cooled reactors than with H₂O cooled reactors, the majority of the heat transfer studies were for D₂O as the coolant. A limited parametric study was made to determine the maximum permissible power densities as a function of several variables. Variables considered were: (1) D₂O velocity, (2) metal/D₂O ratio, (3) flow between flat plates and across rods, (4) channel width and rod diameter, and (5) pressure level (300 psia and 2000 psia).

Hot channel factors were introduced into the parametric study, although it was felt that local boiling over a small area is not serious and could be tolerated if the heat flux is well below the burnout point. However, since the calculated power densities in the non-boiling region for flat plates were adequate, boiling heat transfer studies were not pursued extensively. The methods used in the heat transfer calculations are given in Appendix 4.0.

4.1 Results and Discussion

4.1.1 Flat Plates

The maximum permissible power densities necessary to maintain wall temperatures at or below the saturation temperature (including hot channel factors) for 300 psia and 2000 psia pressure levels with flat plates are shown in Figures 4.A and 4.B, for water velocities of 30 and 20 ft/sec, respectively, as functions of channel width and metal/D₂O ratios. Cross plots on the curves are shown for 0.050-, 0.060-, and 0.070-inch thick plates. The graphs indicate that the plates should be as thin as practical (limited by mechanical considerations) in order to maximize the permissible power density (for any given metal/D₂O ratio). It can also be noted that each plate size has an optimum metal/D₂O ratio from heat transfer considerations. For the 0.060-inch thick plate, it occurs at about a metal/D₂O ratio of 1.0. Since the overall problem is to maximize the neutron flux in the test hole, the heat transfer capabilities for a given size plate need to be compared to the nuclear characteristics of the reactor before finally optimizing the metal/D₂O ratio.

Check calculations were made for a light water cooled core. The values of $Q_{\max}$ obtained were about 90% of the values for D_2O .

A check was also made of the effect of increasing $\bar{H}$ (extrapolated length of reactor) from 54.11 to 58.0 inches, for a fluid velocity of 30 ft/sec, channel width of 0.060 inches and metal/ D_2O ratio of 1.0. The difference in $Q_{\max}$ was negligible ($< 1\%$).

The pressure drop across the core is shown in Figure 4.C for D_2O velocities of 30 and 20 ft/sec. For a 0.060 inch channel and 30 ft/sec flow velocity, the pressure drop across the reactor is about 77 psia. In order to maintain the maximum wall temperature below the saturation temperature for the $Q_{\max}$ indicated, the inlet pressure must be increased above the pressures used in this report by about

$$\Delta P \frac{z - \bar{R}}{L} \text{ psia,}$$

where $\bar{R}$ is the distance between the reactor face and the extrapolated length to zero flux, z is the distance to the hot spot measured from $R = 0$, and L is the length of the reactor (36 inches).

The ΔP curve indicates that pumping power will increase sharply with decreasing channel width. However, the pumping power probably should be subjugated (within reasonable limits) to achievement of desired nuclear characteristics in the test hole.

The average bulk temperature rise for 1 MW/liter average power density is shown in Figure 4.D. To obtain the mixed mean bulk temperature out for a given reactor, values given should be multiplied by the actual power density in MW/liter.

Heat transfer burnout calculations were made for 0.060-inch and 0.070-inch channels with D_2O velocities of 20 and 30 ft/sec. The following table summarizes the results.

Table 4.1.1 Results of Burnout Calculations

<u>V</u> <u>ft/sec</u>	<u>D_e</u> <u>inches</u>	<u>T_b</u> <u>°F</u>	<u>Q</u> <u>MW/liter</u>	
			<u>300 psia</u>	<u>2000 psia</u>
30	0.060	300	15.7	26.8
30	0.060	200	23.6	34.6
20	0.060	300	11.0	18.6
20	0.600	200	16.5	24.0
30	0.070	300	14.5	24.4
30	0.070	200	21.5	31.9
20	0.070	300	10.2	17.4
20	0.070	200	15.4	22.5

The equation used for obtaining the above may predict somewhat high values for Q. However, the calculated burnout densities are about ten times those needed in the test reactor cores and thus the margin of safety is certainly ample.

4.1.2 Flow Across Rods

Flow across rods has some obvious mechanical, fabrication, and operational advantages over flat plates. However, although the heat transfer coefficients are very high, the small spacing required to give the desired metal/D₂O ratios limit the weight flow rate of D₂O and therefore the maximum permissible power densities are considerably lower than those for flat plates (with the same pressure drop across the core).

The maximum permissible power densities are shown in Figure 4.E and 4.F for maximum velocities of 15 and 10 ft/sec, respectively. The values are probably 10% - 20% conservative since mixing between channels was neglected. Although the values for Q_{max} are probably too low (for operation in the non-boiling region) to attain the required neutron flux in the test hole at 300 psia pressure level, operation in the local boiling region might be permissible and thereby the desired nuclear characteristics might be attained. No channel blocking should occur in this type element with local boiling since all channels are interconnected. Future studies probably should evaluate the desirability of operating this type element in the

boiling region.

Pressure drops across the core are shown in Figure 4.G as a function of rod diameter and maximum D₂O velocity. The average bulk temperature rise for 1 MW/liter power density is shown in Figure 4.H.

APPENDIX 1.0

TEST FACILITY REQUIREMENTS FOR THE ADVANCED ENGINEERING TEST REACTOR

Two - 2000 psi (500 - 600°F) circulating light water loops.

one - 3 inch I.D.

one - 4 inch I.D.

Unperturbed flux, 10^{15} n/cm²sec

Perturbed flux, 4 - 5×10^{14} n/cm²sec

The flux should be uniform over the cross section of the loop and exhibit 50% uniformity over a 2.5 to 3 foot section of length. The in-pile section of the loop should be vertical.

The loop should be designed for 6000 hours operating, capable of removing 5 MW of heat, and allow fuel element testing to destruction

Three - 6 inch x 6 inch versatile loop space

One - 4 inch x 4 inch versatile loop space

Unperturbed flux, 1.5×10^{15}

These facilities should be flexible enough to allow insertion of either gas or liquid metal cooled loops. The gas cooled loop should operate at an average temperature of 2200°F and up to 300 psi. The liquid metal loops should operate in the order of 1500°F average temperature and up to 300 psi.

One - 3 inch diameter liquid metal loop

Unperturbed flux, 10^{15} n/cm²sec

The loop should be designed to operate up to 300 psi and have an average temperature of 1500°F.

In addition, 35 - 40 spaces for capsule irradiation, some of which are in the highest flux region, are required.

Component testing requiring 10^{16} - 10^{18} n/cm²sec flux is really outside the scope of the AETR, but large irradiation facilities at any flux attainable can be used.

APPENDIX 2.0

TABULATION OF REACTOR SURVEY RESULTS

On the following pages are summarized the results of the reactor studies. The problems are identified by three numbers and a letter, as follows:

X.YZL

where X is the problem number, which is the same for reactors of similar core dimensions and overall size.

Y = 0 for D₂O cooled reactor cores and 5 for H₂O cooled reactor cores

Z = 1, 2, 3, or 4 for reactor core metal-to-water ratios of 0.7, 1.0, 1.3, and 1.6 respectively.

L = a for no test section in test hole
 b for T-1 test section in test hole
 c for T-2 test section in test hole
 d for T-3 test section in test hole

T-1 and T-2 are water-cooled test sections.
 T-3 is an air-cooled test section

The general survey problems are listed first, while those for specific reactor arrangements are listed last. The general survey involved test holes filled either with H₂O, D₂O, or water-cooled test assemblies. Specific reactor arrangements were studied for reactors with air-cooled and water-cooled test loops. The water-cooled loop reactors are problem numbers 29 to 32 inclusive. The gas-cooled loop reactors are problem numbers 33 to 37 inclusive.

All reactor studies were based on 3 foot long cores with an extrapolated length of 137.44 cm, which gave a longitudinal maximum-to-average neutron flux ratio of 1.21 assuming a chopped cosine distribution. Subsequent calculations showed that for most cases the extrapolated length is considerably longer (~170 cm.) and thus the longitudinal maximum-to-average is smaller than that used.

The T-1 test section used for determining the perturbed neutron flux in the water cooled loop is a typical water-cooled fuel assembly with a U²³⁵ loading of 0.77×10^{20} atoms/cc. The

T-2 test section has twice the fuel loading. The T-3 test section is a typical air-cooled test section with 6×10^{20} atoms U^{235} per cc.

The reactor power and the associated power densities necessary to achieve the required unperturbed neutron fluxes in the test region were given most weight in selecting reactor arrangements for more detailed study.

Problem Number	1.01a	1.01b	2.01a	2.01b	2.01c	3.01a	3.01b
*Region 1; R_O , cm; Comp.	4 D ₂ O	4 T-1	4 D ₂ O	4 T-1	4 T-2	4 D ₂ O	4 T-1
Region 2; R_O , cm; Comp.	24 D ₂ O	24 D ₂ O	24 D ₂ O	24 D ₂ O	24 D ₂ O	40 D ₂ O	40 D ₂ O
Region 3; R_O , cm; Comp.	39 Core	39 Core	32 Core	32 Core	32 Core	55 Core	55 Core
Region 4; R_O , cm; Comp.	69 D ₂ O	69 D ₂ O	62 D ₂ O	62 D ₂ O	62 D ₂ O	130 D ₂ O	130 D ₂ O
Region 5; R_O , cm; Comp.							
**Reactor Core							
Volume, liters	271.5	271.5	123	123	123	409	409
Kg U ²³⁵	5.32	5.32	4.82	4.82	4.82	8.02	8.02
M/W ratio	0.7	0.7	0.7	0.7	0.7	0.7	0.7
Coolant	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
$N_U \cdot 10^{20}$ atoms U ²³⁵ /cc	0.5	0.5	1	1	1	0.5	0.5
k	1.189	1.213	1.230	1.255	1.267	1.323	1.335
Radial max/av power in core in test	1.444	1.471 1.084	1.246	1.249 1.079	1.247 1.177	1.346	1.357 1.125
Radial max/av ϕ_{TH} in test region	1.007		1.007			1.001	
***Maximum power density in core, MW/liter per 100 MW	0.644	0.656	1.23	1.23	1.23	0.398	0.401
Av. ϕ_{TH} in test region, 10^{13} n/cm ² sec per MW	1.582	1.237	1.753	1.360	1.042	0.996	0.728
Reactor power for 5×10^{14} ϕ_{TH} av. in test, MW		40.4		36.8	48.0		68.7
1×10^{15} ϕ_{TH} av. in regional, MW	63.2		57.0			100	
5 MW in test, MW		74.6		83.3	53.1		162
Maximum core power den- sity for 5×10^{14} ϕ_{TH} av. in test, MW/lit		0.265		0.453	0.590		0.275

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	4.01a	4.01b	5.01a	5.01b	5.03a	5.03b	5.04a
*Region 1; R_0 , cm; Comp.	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O
Region 2; R_0 , cm; Comp.	6 H ₂ O	6 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O
Region 3; R_0 , cm; Comp.	22 Core	22 Core	24 Core	24 Core	24 Core	24 Core	24 Core
Region 4; R_0 , cm; Comp.	52 D ₂ O	52 D ₂ O	54 D ₂ O	54 D ₂ O	54 D ₂ O	54 D ₂ O	54 D ₂ O
Region 5; R_0 , cm; Comp.							
**Reactor Core							
Volume, liters	128.6	128.6	147	147	147	147	147
Kg U ²³⁵	2.52	2.52	2.88	2.88	2.88	2.88	2.88
M/W ratio	0.7	0.7	0.7	0.7	1.3	1.3	1.6
Coolant	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
$N_U \cdot 10^{20}$ atoms U ²³⁵ /cc	0.5	0.5	0.5	0.5	0.5	0.5	0.5
k	1.006	1.079	1.002	1.085	0.901	0.990	0.866
Radial max/av power in core in test	1.452	1.420 1.006	1.459	1.514 1.060	1.445	1.505 1.054	1.440
Radial max/av ϕ_{TH} in test region	1.143		1.057		1.068		1.072
***Maximum power density in core, MW/liter per 100 MW	1.37	1.34	1.2	1.25	1.19	1.24	1.185
Av. ϕ_{TH} in test region, 10^{13} n/cm ² sec per MW	3.26	2.17	3.25	2.38	3.73	2.74	3.94
Reactor power for 5×10^{14} ϕ_{TH} av in test, MW		23.0		21.0		18.3	
1×10^{15} ϕ_{TH} av. in regional, MW	30.8		30.8		26.8		25.4
5 MW in test, MW		50.3		47.6		41.3	
Maximum core power den- sity for 5×10^{14} ϕ_{TH} av. in test, MW/lit		0.309		0.263		0.227	

* R_0 - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	5.04b	5.54a	5.54b	6.01a	6.01b	7.01a	7.01b
*Region 1; R_o , cm; Comp.	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1
Region 2; R_o , cm; Comp.	8 H ₂ O	8 H ₂ O	8 H ₂ O	12 H ₂ O	12 H ₂ O	6 H ₂ O	6 H ₂ O
Region 3; R_o , cm; Comp.	24 Core	24 Core	24 Core	28 Core	28 Core	16 Core	16 Core
Region 4; R_o , cm; Comp.	54 D ₂ O	54 D ₂ O	54 D ₂ O	58 D ₂ O	58 D ₂ O	46 D ₂ O	46 D ₂ O
Region 5; R_o , cm; Comp.							
**Reactor Core							
Volume, liters	147	147	147	184	184	63.2	63.2
Kg U ²³⁵	2.88	2.88	2.88	3.60	3.60	2.48	2.48
M/W ratio	1.6	1.6	1.6	0.7	0.7	0.7	0.7
Coolant	D ₂ O	H ₂ O	H ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
$N_U \cdot 10^{20}$ atoms U ²³⁵ /cc	0.5	0.5	0.5	0.5	0.5	1	1
k	0.958	0.924	1.110	0.974	1.030	1.030	1.109
Radial max/av power in core	1.500	1.324	1.329	1.406	1.480	1.420	1.363
in test	1.051		1.055		1.097		1.009
Radial max/av ϕ_{TH} in test region		1.057		1.002		1.168	
***Maximum power density in core, MW/liter per 100 MW							
	1.235	1.09	2.15	0.925	0.973	2.72	2.61
Av. ϕ_{TH} in test region, 10^{13} n/cm ² sec per MW	2.89	2.56	2.18	2.31	1.91	3.85	2.45
Reactor power for 5×10^{14} ϕ_{TH} av. in test, MW	17.3		22.9		26.2		20.4
1×10^{15} ϕ_{TH} av. in regional, MW		39.1		43.2		26.0	
5 MW in test, MW	39.0		51.7		60.8		43.9
Maximum core power den- sity for 5×10^{14} ϕ_{TH} av. in test, MW/lit	0.214		0.492		0.262		0.532

* R_o - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	7.01b	7.01c	8.01a	8.01b	8.01c	8.03a	8.03b
*Region 1; R_O , cm; Comp.	4 T-1	4 T-2	4 H ₂ O	4 T-1	4 T-2	4 H ₂ O	4 T-1
Region 2; R_O , cm; Comp.	6 H ₂ O	6 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O
Region 3; R_O , cm; Comp.	16 Core	16 Core	18 Core	18 Core	18 Core	18 Core	18 Core
Region 4; R_O , cm; Comp.	21 Void	46 D ₂ O	48 D ₂ O	48 D ₂ O	48 D ₂ O	48 D ₂ O	48 D ₂ O
Region 5; R_O , cm; Comp.	46 D ₂ O						
**Reactor Core							
Volume, liters	63.2	63.2	74.7	74.7	74.7	74.7	74.7
Kg U ²³⁵	2.48	2.48	2.93	2.93	2.93	2.93	2.93
M/W ratio	0.7	0.7	0.7	0.7	0.7	1.3	1.3
Coolant	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
N _U -10 ²⁰ atoms U ²³⁵ /cc	1	1	1	1	1	1	1
k	0.901	1.124	1.033	1.128	1.151	0.962	1.062
Radial max/av power in core		1.304	1.443	1.464	1.439	1.409	1.430
in test		1.069		1.053	1.142		1.049
Radial max/av ϕ_{TH} in test region			1.071			1.078	
***Maximum power density in core, MW/liter per 100 MW							
		2.50	2.34	2.37	2.33	2.78	2.32
Av. ϕ_{TH} in test region, 10 ¹³ n/cm ² sec per MW		1.80	3.93	2.74	1.96	4.28	2.98
Reactor power for 5x10 ¹⁴ ϕ_{TH} av. in test, MW		27.8		18.2	25.5		16.8
1x10 ¹⁵ ϕ_{TH} av. in regional, MW			25.4			23.4	
5 MW in test, MW		28.6		41.0	27.6		37.6
Maximum core power den- sity for 5x10 ¹⁴ ϕ_{TH} av. in test, MW/lit		0.695		0.431	0.594		0.390

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	8.04a	8.04b	8.54a	8.54b	9.01a	9.01b	9.01c
*Region 1; R_0 , cm; Comp.	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 T-2
Region 2; R_0 , cm; Comp.	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	10 H ₂ O	10 H ₂ O	10 H ₂ O
Region 3; R_0 , cm; Comp.	18 Core	18 Core	18 Core	18 Core	20 Core	20 Core	20 Core
Region 4; R_0 , cm; Comp.	48 D ₂ O	48 D ₂ O	48 D ₂ O	48 D ₂ O	50 D ₂ O	50 D ₂ O	50 D ₂ O
Region 5; R_0 , cm; Comp.							
**Reactor Core							
Volume, liters	74.7	74.7	74.7	74.7	86.2	86.2	86.2
Kg U ²³⁵	2.93	2.93	2.93	2.93	3.38	3.38	3.38
M/W ratio	1.6	1.6	1.6	1.6	0.7	0.7	0.7
Coolant	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
$N_U \cdot 10^{20}$ atoms U ²³⁵ /cc	1	1	1	1	1	1	1
	0.948	1.050	1.032	1.110	1.019	1.107	1.133
Radial max/av power in core	1.398	1.418	1.304	1.329	1.433	1.482	1.480
in test		1.047		1.055		1.079	1.179
Radial max/av ϕ_{TH} in test region	1.080		1.071		1.027		
***Maximum power density in core, MW/liter per 100 MW	2.26	2.30	2.11	2.15	2.01	2.08	2.08
Av. ϕ_{TH} in test region, 10^{15} n/cm ² sec per MW	4.41	3.06	3.13	2.18	3.49	2.60	1.85
Reactor power for 5×10^{14} ϕ_{TH} av. in test, MW		16.3		22.9		19.2	27.0
1×10^{15} ϕ_{TH} av. in regional, MW	22.7		31.9		28.7		
5 MW in test, MW		36.7		51.7		43.9	29.9
Maximum core power den- sity for 5×10^{14} ϕ_{TH} av. in test, MW/lit		0.376		0.492		0.399	0.562

* R_0 - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	10.01a	10.01b	10.01c	11.01a	11.01b	11.03a	11.03b
*Region 1; R_0 , cm; Comp.	4 H ₂ O	4 T-1	4 T-2	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1
Region 2; R_0 , cm; Comp.	12 H ₂ O	12 H ₂ O	12 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O
Region 3; R_0 , cm; Comp.	22 Core	22 Core	22 Core	12 Core	12 Core	12 Core	12 Core
Region 4; R_0 , cm; Comp.	52 D ₂ O	52 D ₂ O	52 D ₂ O	42 D ₂ O	42 D ₂ O	42 D ₂ O	42 D ₂ O
Region 5; R_0 , cm; Comp.							
**Reactor Core							
Volume, liters	97.7	97.7	97.7	31.0	31.0	31.0	31.0
Kg U ²³⁵	3.83	3.83	3.83	2.43	2.43	2.43	2.43
M/W ratio	0.7	0.7	0.7	0.7	0.7	1.3	1.3
Coolant	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
N _U -10 ²⁰ atoms U ²³⁵ /cc	1	1	1	2	2	2	2
k	1.002	1.071	1.096	1.041	1.123	0.993	1.076
Radial max/av power in core in test	1.414	1.465 1.093	1.474 1.196	1.363	1.298 1.023	1.331	1.269 1.029
Radial max/av ϕ_{TH} in test region	1.002			1.188		1.197	
***Maximum power density in core, MW/liter per 100 MW	1.75	1.81	1.83	5.32	5.07	5.20	4.95
Av. ϕ_{TH} in test region, 10 ¹³ n/cm ² sec per MW	2.87	2.26	1.62	4.38	2.71	4.56	2.81
Reactor power for 5x10 ¹⁴ ϕ_{TH} av. in test, MW		22.1	30.9		18.45		17.8
1x10 ¹⁵ ϕ_{TH} av. in regional, MW	34.8			22.8		21.9	
5 MW in test, MW		51.0	34.7		39.7		38.3
Maximum core power den- sity for 5x10 ¹⁴ ϕ_{TH} av. in test, MW/lit		0.400	0.565		0.935		0.881

* R_0 - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	12.01a	12.01b	12.03a	12.03b	12.04a	12.04b	12.54a
*Region 1; R_O , cm; Comp.	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O
Region 2; R_O , cm; Comp.	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O
Region 3; R_O , cm; Comp.	14 Core	14 Core	14 Core	14 Core	14 Core	14 Core	14 Core
Region 4; R_O , cm; Comp.	44 D ₂ O	44 D ₂ O	44 D ₂ O	44 D ₂ O	44 D ₂ O	44 D ₂ O	44 D ₂ O
Region 5; R_O , cm; Comp.							
**Reactor Core							
Volume, liters	37.9	37.9	37.9	37.9	37.9	37.9	37.9
Kg U ²³⁵	2.97	2.97	2.97	2.97	2.97	2.97	2.97
M/W ratio	0.7	0.7	1.3	1.3	1.6	1.6	1.6
Coolant	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	H ₂ O
$N_U \cdot 10^{20}$ atoms U ²³⁵ /cc	2	2	2	2	2	2	2
k	1.051	1.156	1.005	1.113	0.996	1.104	1.077
Radial max/av power in core in test	1.393	1.392 1.049	1.364	1.365 1.045	1.351	1.356 1.044	1.259
Radial max/av ϕ_{TH} in test region	1.081		1.086		1.087		1.082
***Maximum power density in core, MW/liter per 100 MW	4.45	4.44	4.35	4.36	4.31	4.33	4.02
Av. ϕ_{TH} in test region, 10^{13} n/cm ² sec per MW	4.507	3.045	4.768	3.217	4.84	3.26	3.78
Reactor power for 5×10^{14} ϕ_{TH} av. in test, MW		16.42		15.5		15.34	
1×10^{15} ϕ_{TH} av. in regional, MW	22.2		21.0		20.7		26.5
5 MW in test, MW		37.0		34.8		34.5	
Maximum core power den- sity for 5×10^{14} ϕ_{TH} av. in test, MW/lit		0.729		0.676		0.664	

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	12.54b	13.01a	13.01b	13.03a	13.03b	14.01a	14.01b
*Region 1; R_O , cm; Comp.	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1
Region 2; R_O , cm; Comp.	8 H ₂ O	10 H ₂ O	10 H ₂ O	10 H ₂ O	10 H ₂ O	12 H ₂ O	12 H ₂ O
Region 3; R_O , cm; Comp.	14 Core	16 Core	16 Core	16 Core	16 Core	18 Core	18 Core
Region 4; R_O , cm; Comp.	44 D ₂ O	46 D ₂ O	46 D ₂ O	46 D ₂ O	46 D ₂ O	48 D ₂ O	48 D ₂ O
Region 5; R_O , cm; Comp.							
**Reactor Core							
Volume, liters	37.9	44.8	44.8	44.8	44.8	51.7	51.7
Kg U ²³⁵	2.97	3.51	3.51	3.51	3.51	4.05	4.05
M/W ratio	1.6	0.7	0.7	1.3	1.3	0.7	0.7
Coolant	H ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
$N_U \cdot 10^{20}$ atoms U ²³⁵ /cc	2	2	2	2	2	2	2
k	1.168	1.039	1.138	0.993	1.097	1.020	1.101
Radial max/av power in core	1.263	1.393	1.420	1.367	1.394	1.383	1.416
in test	1.048		1.077		1.068		1.091
Radial max/av ϕ_{TH} in test region		1.033		1.036		1.007	
***Maximum power density in core, MW/liter per 100 MW	4.03	3.76	3.84	3.69	3.77	3.24	3.31
Av. ϕ_{TH} in test region, 10^{13} n/cm ² sec per MW	2.54	4.02	2.907	4.317	3.12	3.35	2.55
Reactor power for 5×10^{14} ϕ_{TH} av. in test, MW	19.7		17.2		16.0		19.6
1×10^{15} ϕ_{TH} av. in regional, MW		24.9		23.2		29.9	
5 MW in test, MW	44.2		39.4		36.6		45.0
Maximum core power den- sity for 5×10^{14} ϕ_{TH} av. in test, MW/lit	0.794		0.660		0.603		0.649

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	14.03a	14.03b	15.01a	15.01b	16.01a	16.01b	16.03a
*Region 1; R_O , cm; Comp.	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O
Region 2; R_O , cm; Comp.	12 H ₂ O	12 H ₂ O	6 H ₂ O	6 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O
Region 3; R_O , cm; Comp.	18 Core	18 Core	9.5 Core	9.5 Core	11.5 Core	11.5 Core	11.5 Core
Region 4; R_O , cm; Comp.	48 D ₂ O	48 D ₂ O	39.5 D ₂ O	39.5 D ₂ O	41.5 D ₂ O	41.5 D ₂ O	41.5 D ₂ O
Region 5; R_O , cm; Comp.							
**Reactor Core							
Volume, liters	51.7	51.7	15.6	15.6	19.6	19.6	19.6
Kg U ²³⁵	4.05	4.05		2.446		3.07	3.07
M/W ratio	1.3	1.3	0.7	0.7	0.7	0.7	1.3
Coolant	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
$N_U \cdot 10^{20}$ atoms U ²³⁵ /cc	2	2	4	4	4	4	4
k	0.975	1.060	1.042	1.125	1.059	1.170	1.032
Radial max/av power in core	1.359	1.393	1.328	1.26	1.362	1.352	1.314
in test		1.090		1.034		1.044	
Radial max/av ϕ_{TH} in test region	1.009		1.201		1.088		1.092
***Maximum power density in core, MW/liter per 100 MW	3.18	3.26	10.3	9.77	8.41	8.35	3.11
Av. ϕ_{TH} in test region, 10^{13} n/cm ² sec per MW	3.61	2.77	4.78	2.902	4.92	3.27	5.08
Reactor power for 5×10^{14} ϕ_{TH} av. in test, MW		18.0		17.23		15.3	
1×10^{15} ϕ_{TH} av. in regional, MW	27.7		20.9		20.3		19.7
5 MW in test, MW		41.7		36.98		34.3	
Maximum core power den- sity for 5×10^{14} ϕ_{TH} av. in test, MW/lit		0.587		1.68		1.28	

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	16.03b	16.04a	16.04b	16.51b	16.52b	16.53b	16.54a
*Region 1; R_O , cm; Comp.	4 T-1	4 H ₂ O	4 T-1	4 T-1	4 T-1	4 T-1	4 H ₂ O
Region 2; R_O , cm; Comp.	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O	8 H ₂ O
Region 3; R_O , cm; Comp.	11.5 Core	11.5 Core	11.5 Core	11.5 Core	11.5 Core	11.5 Core	11.5 Core
Region 4; R_O , cm; Comp.	41.5 D ₂ O	41.5 D ₂ O	41.5 D ₂ O	41.5 D ₂ O	41.5 D ₂ O	41.5 D ₂ O	41.5 D ₂ O
Region 5; R_O , cm; Comp.							
**Reactor Core							
Volume, liters	19.6	19.6	19.6	19.6	19.6	19.6	19.6
Kg U ²³⁵	3.07	3.07	3.07	3.07	3.07	3.07	3.07
M/W ratio	1.3	1.6	1.6	0.7	1.0	1.3	1.6
Coolant	D ₂ O	D ₂ O	D ₂ O	H ₂ O	H ₂ O	H ₂ O	H ₂ O
$N_U \cdot 10^{20}$ atoms U ²³⁵ /cc	4	4	4	4	4	4	4
k	1.144	1.028	1.141	1.250	1.235	1.224	1.090
Radial max/av power in core	1.306	1.301	1.292	1.255	1.243	1.231	1.206
in test	1.042		1.042	1.048	1.047	1.046	
Radial max/av ϕ_{TH} in test region		1.092					1.090
***Maximum power density in core, MW/liter per 100 MW	8.06	8.03	7.98	7.75	7.67	7.60	7.45
Av. ϕ_{TH} in test region, 10^{13} n/cm ² sec per MW	3.35	5.11	3.375	2.508	2.608	2.684	4.33
Reactor power for 5×10^{14} ϕ_{TH} av. in test, MW	14.9		14.81	19.9	19.2	18.6	
1×10^{15} ϕ_{TH} av. in regional, MW		19.6					23.1
5 MW in test, MW	33.1		33.2	44.8	43.0	41.8	
Maximum core power den- sity for 5×10^{14} ϕ_{TH} av. in test, MW/lit	1.20		1.182	1.542	1.470	1.416	

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	16.54a	17.01a	17.01b	18.01a	18.01b	19.01a	19.01b
*Region 1; R _O , cm; Comp.	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1
Region 2; R _O , cm; Comp.	8 H ₂ O	10 H ₂ O	10 H ₂ O	12 H ₂ O	12 H ₂ O	6 H ₂ O	6 H ₂ O
Region 3; R _O , cm; Comp.	11.5 Core	13.5 Core	13.5 Core	15.5 Core	15.5 Core	8 D ₂ O	8 D ₂ O
Region 4; R _O , cm; Comp.	41.5 D ₂ O	43.5 D ₂ O	43.5 D ₂ O	45.5 D ₂ O	45.5 D ₂ O	18 Core	18 Core
Region 5; R _O , cm; Comp.						48 D ₂ O	48 D ₂ O
**Reactor Core							
Volume, liters	19.6	23.6	23.6	27.7	27.7	74.7	74.7
Kg U ²³⁵	3.07	3.70	3.70	4.34	4.34	2.928	2.928
M/W ratio	1.6	0.7	0.7	0.7	0.7	0.7	0.7
Coolant	H ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
N _U -10 ²⁰ atoms U ²³⁵ /cc	4	4	4	4	4	1	1
k	1.191	1.048	1.155	1.029	1.117	1.061	1.137
Radial max/av power in core	1.203	1.368	1.383	1.363	1.387	1.339	1.318
in test	1.044		1.074		1.090		1.013
Radial max/av ϕ_{TH} in test region		1.037		1.010		1.132	
***Maximum power density in core, MW/liter per 100 MW	7.43	7.014	7.09	5.96	6.07	2.17	2.13
Av. ϕ_{TH} in test region, 10 ¹³ n/cm ² sec per MW	2.85	4.432	3.133	3.694	2.770	3.36	2.23
Reactor power for 5x10 ¹⁴ ϕ_{TH} av. in test, MW	17.5		15.96		18.05		22.4
1x10 ¹⁵ ϕ_{TH} av. in regional, MW		22.51		27.1		29.8	
5 MW in test, MW	39.2		36.4		41.6		49.5
Maximum core power den- sity for 5x10 ¹⁴ ϕ_{TH} av. in test, MW/lit	1.300		1.13		1.096		0.477

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	20.01a	20.01b	21.01a	21.01b	22.01a	22.01b	23.01a
*Region 1; R_0 , cm; Comp.	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O
Region 2; R_0 , cm; Comp.	6 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O
Region 3; R_0 , cm; Comp.	10 D ₂ O	10 D ₂ O	12 D ₂ O	12 D ₂ O	14 D ₂ O	14 D ₂ O	8 Void
Region 4; R_0 , cm; Comp.	20 Core	20 Core	22 Core	22 Core	24 Core	24 Core	18 Core
Region 5; R_0 , cm; Comp.	50 D ₂ O	50 D ₂ O	52 D ₂ O	52 D ₂ O	54 D ₂ O	54 D ₂ O	48 D ₂ O
**Reactor Core							
Volume, liters	86.2	86.2	97.7	97.7	109.2	109.2	74.7
Kg U ²³⁵	3.38	3.38	3.83	3.83	4.28	4.28	2.928
M/W ratio	0.7	0.7	0.7	0.7	0.7	0.7	0.7
Coolant	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
N_U -10 ²⁰ atoms U ²³⁵ /cc	1	1	1	1	1	1	1
k	1.091	1.163	1.119	1.186	1.145	1.207	1.024
Radial max/av power in core in test	1.309	1.310 1.030	1.304	1.318 1.044	1.311	1.333 1.057	1.280
Radial max/av ϕ_{TH} in test region	1.103		1.078		1.056		1.176
***Maximum power density in core, MW/liter per 100 MW	1.84	1.84	1.61	1.63	1.45	1.48	2.07
Av. ϕ_{TH} in test region, 10 ¹³ n/cm ² sec per MW	2.92	2.015	2.551	1.819	2.239	1.644	3.301
Reactor power for 5x10 ¹⁴ ϕ_{TH} av. in test, MW		24.8		27.5		30.4	
1x10 ¹⁵ ϕ_{TH} av. in regional, MW	34.2		39.2		44.7		30.3
5 MW in test, MW		54.9		61.7		68.4	
Maximum core power den- sity for 5x10 ¹⁴ ϕ_{TH} av. in test, MW/lit		0.456		0.448		0.450	

* R_0 - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	23.01b	24.01a	24.01b	25.01a	25.01b	26.01a	26.01b
*Region 1; R _O , cm; Comp.	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1	4 H ₂ O	4 T-1
Region 2; R _O , cm; Comp.	6 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O	6 H ₂ O
Region 3; R _O , cm; Comp.	8 Void	10 Void	10 Void	12 Void	12 Void	14 Void	14 Void
Region 4; R _O , cm; Comp.	18 Core	20 Core	20 Core	22 Core	22 Core	34 Core	34 Core
Region 5; R _O , cm; Comp.	18 D ₂ O	50 D ₂ O	50 D ₂ O	52 D ₂ O	52 D ₂ O	54 D ₂ O	54 D ₂ O
**Reactor Core							
Volume, liters	74.7	86.2	86.2	97.7	97.7	109.2	109.2
Kg U ²³⁵	2.928	3.38	3.38	3.83	3.83	4.28	4.28
M/W ratio	0.7	0.7	0.7	0.7	0.7	0.7	0.7
Coolant	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O	D ₂ O
N _{U-10²⁰} atoms U ²³⁵ /cc	1	1	1	1	1	1	1
k	1.089	1.012	1.067	0.997	1.044	0.981	1.021
Radial max/av power in core	1.230	1.248	1.260	1.288	1.299	1.324	1.332
in test	1.015		1.020		1.022		1.024
Radial max/av ϕ_{TH} in test region		1.182		1.186		1.188	
***Maximum power density in core, MW/liter per 100 MW	1.99	1.75	1.77	1.60	1.61	1.47	1.48
Av. ϕ_{TH} in test region, 10 ¹³ n/cm ² sec per MW	2.072	2.883	1.793	2.554	1.578	2.288	1.405
Reactor power for 5x10 ¹⁴ ϕ_{TH} av. in test, MW	24.1		27.9		31.7		35.6
1x10 ¹⁵ ϕ_{TH} av. in regional, MW		34.7		39.2		43.7	
5 MW in test, MW	52.1		60.1		68.0		76.3
Maximum core power den- sity for 5x10 ¹⁴ ϕ_{TH} av. in test, MW/lit	0.480		0.494		0.510		0.527

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	27.01a	28.01a					
*Region 1; R_O , cm; Comp.	8 D ₂ O	20 D ₂ O					
Region 2; R_O , cm; Comp.	12 H ₂ O	24 H ₂ O					
Region 3; R_O , cm; Comp.	15.5 Core	32 Core					
Region 4; R_O , cm; Comp.	45.5 D ₂ O	62 D ₂ O					
Region 5; R_O , cm; Comp.							
**Reactor Core							
Volume, liters	27.7	123					
Kg U ²³⁵	4.34	5.32					
M/W ratio	0.7	0.7					
Coolant	D ₂ O	D ₂ O					
$N_U \cdot 10^{20}$ atoms U ²³⁵ /cc	4	1					
k	1.166	1.081					
Radial max/av power in core in test	1.412	1.379					
Radial max/av ϕ_{TH} in test region	1.010	1.003					
***Maximum power density in core, MW/liter per 100 MW	6.18	1.36					
Av. ϕ_{TH} in test region, 10^{13} n/cm ² sec per MW	3.55	2.66					
Reactor power for 5×10^{14} ϕ_{TH} av. in test, MW							
1×10^{15} ϕ_{TH} av. in regional, MW	28.1	37.6					
5 MW in test, MW							
Maximum core power den- sity for 5×10^{14} ϕ_{TH} av. in test, MW/lit							

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	29.04a	29.04b	29.52b	30.52b	31.52b	32.52b
*Region 1; R _O , cm; Comp.	5 H ₂ O	5 T-1	5 T-1	5 T-1	5 T-1	5 T-1
Region 2; R _O , cm; Comp.	9 H ₂ O	9 H ₂ O	9 H ₂ O	9 H ₂ O	9 H ₂ O	9 H ₂ O
Region 3; R _O , cm; Comp.	9.6 Zr	9.6 Zr	9.6 Zr	9.6 Zr	9.6 Zr	9.6 Zr
Region 4; R _O , cm; Comp.	10.4 Void	10.4 Void	10.4 Void	10.4 Void	10.4 Void	10.4 Void
Region 5; R _O , cm; Comp.	11 Al	11 Al	11 Al	11 Al	11 Al	11 Al
Region 6; R _O , cm; Comp.	15 Core	15 Core	15 Core	15 Core	15 Core	15 Core
Region 7; R _O , cm; Comp.	18 Al	18 Al	18 Al	18 Al	18 Al	18 Al
Region 8; R _O , cm; Comp.	48 D ₂ O	48 D ₂ O	48 D ₂ O	33 D ₂ O	63 D ₂ O	78 D ₂ O
**Reactor Core						
Volume, liters	29.9	29.9	29.9	29.9	29.9	29.9
Kg U ²³⁵	4.68	4.68	4.68	4.68	4.68	4.68
M/W ratio	1.6	1.6	1.0	1.0	1.0	1.0
Coolant	D ₂ O	D ₂ O	H ₂ O	H ₂ O	H ₂ O	H ₂ O
N _U -10 ²⁰ atoms U ²³⁵ /cc	4	4	4	4	4	4
k	0.957	1.1057	1.183	1.028	1.239	1.263
Radial max/av power in core	1.335	1.334	1.288	1.439	1.241	1.222
in test		1.051	1.059	1.055	1.061	1.061
Radial max/av ϕ_{TH} in test region	1.107					
***Maximum power density in core, MW/liter per 100 MW	5.40	5.44	5.21	5.82	5.02	4.95
Av. ϕ_{TH} in test region, 10 ¹³ n/cm ² sec per MW	4.49	2.91	2.134	2.567	2.0	1.951
Reactor power for 5x10 ¹⁴ ϕ_{TH} av. in test, MW		17.2	23.4	19.5	25	25.6
1x10 ¹⁵ ϕ_{TH} av. in region 1, MW	22.3					
5 MW in test, MW		24.5	33.5	27.7	35.6	36.5
Maximum core power den- sity for 5x10 ¹⁴ ϕ_{TH} av. in test, MW/lit.		0.935	1.219	1.135	1.255	1.267

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-1, T-2, and H₂O at 500°F; T-1 test section U:H₂O:Zr = 1:168:338; T-2 test section U:H₂O:Zr = 1:84:169 (atom).

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	33.02a	33.02d	33.52a	33.52d	34.02a
*Region 1; R_O , cm; Comp.	5 Void	5 T-3	5 Void	5 T-3	5 Void
Region 2; R_O , cm; Comp.	7 Void	7 Void	7 Void	7 Void	7 Void
Region 3; R_O , cm; Comp.	7.2 SS	7.2 SS	7.2 SS	7.2 SS	7.2 SS
Region 4; R_O , cm; Comp.	8 Void	8 Void	8 Void	8 Void	8 Void
Region 5; R_O , cm; Comp.	8.8 Al	8.8 Al	8.8 Al	8.8 Al	8.8 Al
Region 6; R_O , cm; Comp.	12.8 H ₂ O	12.8 H ₂ O	12.8 H ₂ O	12.8 H ₂ O	12.8 H ₂ O
Region 7; R_O , cm; Comp.	13 Al	13 Al	13 Al	13 Al	13 Al
Region 8; R_O , cm; Comp.	17 Core	17 Core	17 Core	17 Core	17 Core
Region 9; R_O , cm; Comp.	18 Al	18 Al	18 Al	18 Al	18 Al
Region 10; R_O , cm; Comp.	48 D ₂ O	48 D ₂ O	48 D ₂ O	48 D ₂ O	38 D ₂ O
**Reactor Core					
Volume, liters	34.5	34.5	34.5	34.5	34.5
Kg U ²³⁵	5.41	5.41	5.41	5.41	5.4
M/W ratio	1.0	1.0	1.0	1.0	1.0
Coolant	D ₂ O	D ₂ O	H ₂ O	H ₂ O	D ₂ O
N _U -10 ²⁰ atoms U ²³⁵ /cc	4	4	4	4	4
k	1.046	1.167	1.131	1.234	0.923
Radial max/av power in core	1.328	1.284	1.224	1.189	1.440
in test		1.167		1.168	
Radial max/av ϕ_{TH} in test region	1.003		1.001		1.002
***Maximum power density in core, MW/liter per 100 MW	4.66	4.50	4.29	4.17	5.05
Av. ϕ_{TH} in test region, 10 ¹³ n/cm ² sec per MW	1.44	0.457	1.131	0.353	1.616
Reactor power for 1.5 x 10 ¹⁵ ϕ_{TH} av. in test region, MW	104		133		92.8
Maximum core power density for 1.5 x 10 ¹⁵ ϕ_{TH} av. in test, MW/lit.	4.86		5.69		4.69

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-3 at 2000°F; H₂O at 500°F; T-3 - fully enriched, SS/UO₂ = 3.75, SS/Void = 0.157.

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

Problem Number	35.02a	35.52a	36.02a	37.02a	37.52a
*Region 1; R_O , cm; Comp.	5 Void	5 Void	5 Void	5 Void	5 Void
Region 2; R_O , cm; Comp.	7 Void	7 Void	7 Void	7 Void	7 Void
Region 3; R_O , cm; Comp.	7.2 SS	7.2 SS	7.2 SS	7.2 SS	7.2 SS
Region 4; R_O , cm; Comp.	8 Void	8 Void	8 Void	8 Void	8 Void
Region 5; R_O , cm; Comp.	8.8 Al	8.8 Al	8.8 Al	8.8 Al	8.8 Al
Region 6; R_O , cm; Comp.	14.8 H ₂ O	14.8 H ₂ O	16.8 H ₂ O	14.8 H ₂ O	14.8 H ₂ O
Region 7; R_O , cm; Comp.	15 Al	15 Al	17 Al	15 Al	15 Al
Region 8; R_O , cm; Comp.	19 Core	19 Core	21 Core	21 Core	21 Core
Region 9; R_O , cm; Comp.	20 Al	20 Al	22 Al	22 Al	22 Al
Region 10; R_O , cm; Comp.	40 D ₂ O	40 D ₂ O	42 D ₂ O	42 D ₂ O	42 D ₂ O
**Reactor Core					
Volume, liters	39.1	39.1	44.8	62.2	62.2
Kg U ²³⁵	6.13	6.13	7.02	9.75	9.75
M/W ratio	1.0	1.0	1.0	1.0	1.0
Coolant	D ₂ O	H ₂ O	D ₂ O	D ₂ O	H ₂ O
N _U -10 ²⁰ atoms U ²³⁵ /cc	4	4	4	4	4
k	0.916	1.016	0.897	0.924	1.142
Radial max/av power in core	1.495	1.336	1.508	1.842	1.512
Radial max/av ϕ_{TH} in test region	1.002	1.002	1.002	1.002	1.002
***Maximum power density in core, MW/liter per 100 MW	4.63	4.13	4.07	3.58	2.94
Av. ϕ_{TH} in test region, 10 ¹³ n/cm ² sec per MW	1.733	1.313	1.559	1.437	0.972
Reactor power for 1.5 x 10 ¹⁵ ϕ_{TH} av. in test region, MW	86.6	114.2	96.2	104.3	154.3
Maximum core power density for 1.5 x 10 ¹⁵ ϕ_{TH} av. in test, MW/lit.	4.00	4.72	4.24	3.73	4.54

* R_O - outer radius of region; Comp. = composition of region; all reactor cores and D₂O at 170°F; T-3 at 2000°F; H₂O at 500°F; T-3 - fully enriched, SS/UO₂ = 3.75, SS/Void = 0.157.

** All cores 91.44 cm high, extrapolated length 137.44 cm.

***Based on longitudinal max/av power ratio of 1.21.

APPENDIX 3.0

NUCLEAR DATA

Preliminary analysis of reactor types suitable for fuel element testing at high flux levels indicated that light and heavy water cooled or moderated reactors would satisfy design specifications. Water moderated reactors may be analyzed with satisfactory precision by three group theory of neutron diffusion. The three group equations which describe the spatial distribution of neutrons in one dimension may be written as follows:

$$\begin{aligned}
 D_F \nabla^2 \phi_F(r) - \left[\Sigma_{S,D,F} + D_F B^2 + \Sigma_{a,F} \right] \phi_F(r) \\
 + \gamma_c \left[\Sigma_{f,F} \phi_F(r) + \Sigma_{f,INT} \phi_{INT}(r) + \Sigma_{f,TH} \phi_{TH}(r) \right] &= 0 \\
 D_{INT} \nabla^2 \phi_{INT}(r) - \left[\Sigma_{S,D,INT} + D_{INT} B^2 + \Sigma_{a,INT} \right] \phi_{INT}(r) \\
 + \Sigma_{S,D,F} \phi_F(r) &= 0 \\
 D_{TH} \nabla^2 \phi_{TH}(r) - \left[D_{TH} B^2 + \Sigma_{a,TH} \right] \phi_{TH}(r) \\
 + \Sigma_{S,D,INT} \phi_{INT}(r) &= 0
 \end{aligned} \tag{1}$$

The first equation describes the neutron balance in the energy range of 10 kev to 10 mev and hence includes all fission neutrons; the second equation covers the range from thermal energy to 10 kev. The last equation describes the neutron balance and spatial distribution of thermal neutrons.

The quantity B^2 is the buckling along the coordinate normal to r . The parametric data discussed in this report were calculated assuming reactors of cylindrical shape so B^2 is the axial buckling of the cylinder

$$B^2 = \pi^2 / L_e^2 \tag{2}$$

L_e is the length of the core plus twice the reflector savings. A value of 23 cm was chosen as the reflector savings for D_2O

moderated reactors; 8.5 cm was used for H₂O moderated reactors¹. These values are for Al/water volume ratios of approximately 0.7. These values were used for all reactor calculations although the Al/water volume ratio was varied and affects the reflector savings.

Equations (1) were solved numerically using an IBM 650 digital calculator. These equations are solved for a critical system and a value of γ_c is computed which sustains criticality. The effective k of a reactor is simply

$$k_{\text{eff}} = \gamma / \gamma_c \quad (3)$$

where $\gamma = 2.46$, the average number of neutrons produced per fission.

The authors are indebted to R. D. Cheverton, et al², for supplying three group constants for Al, D₂O, U(25) and Al-D₂O mixtures for various metal-to-water volume ratios. These data are summarized in Table A3.0b and Figures A3.A to A3.G.

The data for H₂O, S.S. Al-H₂O and Zr-H₂O were computed assuming that the energy dependence of the neutron flux was identical with the MTR neutron flux spectrum³. For example, the diffusion coefficient in the fast group (10 kev - 10 mev), D_F , was calculated by numerically integrating the following expression:

$$D_F = \frac{\int_3^0 \frac{\phi(u)}{3 \Sigma_S (1 - \bar{\mu}_c)} du}{\int_3^0 \phi(u) du} \quad (4)$$

where $u = \log_{10} E_0/E$, $E_0 = 10$ mev, and $\phi(u)$ is the MTR neutron flux spectrum. The slowing down cross section for the fast group is defined by

$$\tau_F = D_F / \Sigma_{S,D,F} \quad (5)$$

so that it is necessary to calculate the partial age in order to find the slowing down cross section.

¹Private communication from J. W. Webster.

²Loco. cit. p 15

³IDO-16133 PPCo, "Reactivity Effect of Reducing the Al/H₂O Ratio in the MTR Core", J. W. Webster.

The partial age for mixtures of H₂O and metal was calculated from empirical formulae¹:

$$\tau_F = \tau_F(\text{H}_2\text{O}) \frac{(1 + \sum_{i=1}^N v_i)^2}{\left[\rho(\text{H}_2\text{O}) + \sum_{i=1}^N v_i y_{F,i} \right] \left[\rho(\text{H}_2\text{O}) + \sum_{i=1}^N v_i x_{F,i} \right]} \quad (6)$$

$\rho(\text{H}_2\text{O})$ is the intrinsic water density; v_i is the metal-to-water ratio for the i th metallic element (where, in general, an alloy of elements is considered rather than a single element); x_i and y_i are empirical constants relevant to the metallic element; and $\tau_F(\text{H}_2\text{O})$ is the partial age for H₂O of unit density in the fast group. Shown below is a table of the empirical constants for various metal elements.

Table A3.0a Empirical Parameters for Metal-to-Water Age Calculations

Element	y_F	x_F	y_{INT}	x_{INT}
Al	0.103	0.922	0.006	0.195
Cr	0.176	1.47	0.014	0.862
Fe	0.186	1.20	0.021	1.39
Ni	0.224	1.62	0.041	2.82
Zr	0.107	1.33	0.005	0.547
C	0.115	0.986	0.046	0.618

The age as given by Equation (6) is considered to be accurate for metal-to-water ratios in the range 0 - 2.0.

The partial age for metallic elements or metallic alloys was computed in the usual fashion. The neutron age from lethargy u to $u_{TH} = 8.5$ is

¹Deutsch, R. W., Nucleonics 15, No. 1, 47 (1957)

$$\tau(u_{TH}) = \int_u^{u_{TH}} \frac{\log_e 10 \, du''}{3 \xi \Sigma_{TR}(u'') \Sigma_S(u'')} \quad (7)$$

Again, $u = \log_{10} E_0/E$, $E_0 = 10 \text{ mev}$.

Equation (7) may be rewritten:

$$\tau(u_{TH}) = \int_{u_1}^{u_{TH}} \frac{\log_e 10 \, du''}{3 \xi \Sigma_{TR}(u'') \Sigma_S(u'')} + \int_u^{u_1} \frac{\log_e 10 \, du''}{3 \xi \Sigma_{TR}(u'') \Sigma_S(u'')} \quad (8)$$

where $u_1 = 3.0$ and $u_1 \geq u \geq 0$. The neutron age from fission neutrons was calculated by weighting the second integral on the right of Equation (8) by the fission spectrum.

$$\begin{aligned} \tau(u_{TH}) = & \int_{u_1}^{u_{TH}} \frac{\log_e 10 \, du''}{3 \xi \Sigma_{TR}(u'') \Sigma_S(u'')} \\ & + \frac{\int_0^{u_1} N(u) \int_u^{u_1} \frac{\log_e 10 \, du''}{3 \xi \Sigma_{TR}(u'') \Sigma_S(u'')} \, du}{\int_0^{u_1} N(u) \, du} \end{aligned} \quad (9)$$

$$\tau(u_{TH}) = \tau_{INT} + \tau_F \quad (10)$$

The intermediate energy diffusion coefficient and slowing down cross section are defined similarly to Equations (4) and (5).

$$D_{INT} = \frac{\int_{8.5}^3 \frac{\phi(u)}{3 \Sigma_S(1 - \bar{\mu}_0)} \, du}{\int_{8.5}^3 \phi(u) \, du} \quad (11)$$

$$\tau_{INT} = \frac{D_{INT}}{\Sigma_{S.D.,INT}} \quad (12)$$

The thermal diffusion coefficient for H_2O has been calculated by Radkowski¹ for various H_2O temperatures. The calculation of D_{TH} for metals or metal alloys is straightforward and D_{TH} for mixtures of these with H_2O may be found from Equation (13):

$$D_{TH}(M - H_2O) = \frac{\left(\frac{v}{1+v}\right) D_{TH}(M) D_{TH}(H_2O)}{D_{TH}(H_2O) + v D_{TH}(M)} \quad (13)$$

v is the metal-to-water volume ratio. Equation (13) may be extended for metal alloys consisting of N kinds of elements.

$$D_{TH}(M-H_2O) = \frac{1}{\left[1 + \sum_{i=1}^N v_i\right] \left[\frac{1}{D_{TH}(H_2O)} + \sum_{i=1}^N \frac{1}{v_i D_{TH,i}}\right]} \quad (14)$$

v_i is the element-water volume ratio for the i^{th} kind of element in the metal alloy.

The calculation of thermal absorption cross section for each element of interest in this study was done by assuming a $1/v$ dependence upon temperature except for U(25), where appropriate corrections for its non- $1/v$ nature were taken into account.

The data calculated by the methods discussed above are summarized in Tables A3.0b, A3.0c, and Figures A3.H to A3.W.

¹Argonne National Laboratory, Quarterly Report, April, May, June, 1950, Physics Division, ANL-4476, July 5, 1950.

Table A3.0c Power Reactor Fuel Elements Nuclear Data

<u>Material</u>	<u>Zr-H₂O (T1)</u>	<u>Zr-H₂O (T2)</u>	<u>SS-Air (T3)</u>
T °F	500	500	2000
$\frac{V(\text{metal})}{V(\text{moderator})}$	1.576	1.576	0.1568
$\frac{V(\text{metal})}{V(\text{fuel})}$	190.8	95.4	3.7515
τ_F cm ²	46.6	46.6	18,670
D _F cm	1.377	1.377	7.602
$\Sigma_{R,F}^*$ cm ⁻¹	0.04033	0.04012	0.006036
$\Sigma_{S.D.F}$ cm ⁻¹	0.0392	0.0392	0.000464
τ_{INT} cm ²	16.2	16.2	6,669
D _{INT} cm	0.842	0.842	2.529
$\Sigma_{R,INT}^*$ cm ⁻¹	0.06323	0.05784	0.04237
$\Sigma_{S.D,INT}$ cm ⁻¹	0.0520	0.0520	0.000379
D _{TH} cm	0.770	0.782	2.527
$\Sigma_{R,TH}^*$ cm ⁻¹	0.0753	0.04223	0.2000
$\Sigma_{f,F}$ cm ⁻¹	0.000312	0.000156	0.001144
$\Sigma_{f,INT}$ cm ⁻¹	0.00830	0.00415	0.03148
$\Sigma_{f,TH}$ cm ⁻¹	0.05552	0.02776	0.1492

$$* \Sigma_R = \Sigma_{S.D.} + \Sigma_a + DB^2$$

$$B^2 = 0.0005225 \text{ cm}^{-2}$$

APPENDIX 4.0

HEAT TRANSFER CALCULATION METHODS

A4.1 Flat Plates

The maximum power density, point of maximum wall temperature, and bulk water temperature rise were obtained from the following three equations (assuming cosine power distributions):

$$Q_{\max} = \frac{t_w - t_{in}}{\frac{1.35(\Delta X + \Delta Y)}{2h} \sin \frac{\pi Z}{\bar{H}} + \frac{1.275(\Delta X + \Delta Y)\bar{H}}{\pi V \rho C_p \Delta X} \left[\cos \frac{\pi(\bar{H} - L)}{2\bar{H}} - \cos \frac{\pi Z}{\bar{H}} \right]} \quad (1)$$

$$Z = \frac{\bar{H}}{\pi} \tan^{-1} \left(- \frac{\pi V \rho C_p \Delta X}{2h\bar{H}} \right) \quad (2)$$

$$(\Delta T_b)_{ave} = \frac{Q_{ave} L(\Delta X + \Delta Y)}{V \rho C_p \Delta X}, \quad (3)$$

where Q is the volumetric heat generation rate, t_w is the maximum permissible wall temperature (saturation temperature); t_{in} is the inlet bulk water temperature; $(\Delta T_b)_{ave}$ is the average bulk D_2O temperature rise; ΔX is the channel width; ΔY is the plate thickness; L is the core length (36 inches); $\bar{H}$ is the total extrapolated length of the core (54.11 inches) to zero flux; Z is the point of maximum wall temperature measured from $\bar{H} = 0$; V is the fluid velocity; h is the heat transfer coefficient; ρ is the fluid density; and C_p is the heat capacity, all in consistent units.

The constants, 1.35 and 1.275, are the hot channel factors for the film and bulk D_2O temperatures, respectively, and are derived in a following chapter.

The heat transfer coefficient, h , was obtained from:

$$\frac{hD_e}{k_f} = 0.020 (N_{Re})_f^{0.8} (N_{Pr})_f^{0.4}, \quad (4)$$

where D_e is the equivalent diameter of the channel, k_f is the

thermal conductivity of the water, and $(N_{Re})_f$ and $(N_{Pr})_f$ are the Reynolds and Prandtl numbers, respectively. All physical properties were evaluated for an average film temperature of 300°F. This gave conservative values for pressures much above 300 psia. Except for density, the physical properties of D₂O were assumed to be the same as those for H₂O.

The points of maximum wall temperature, Z, were calculated with V and ΔX as variables. These values of Z were inserted into equation (1) to obtain Q_{max} as a function of channel width, metal/D₂O ratio and fluid velocity. For each value of ΔX (channel width) and metal/D₂O ratio, a unique value of ΔY (plate thickness) was obtained from:

$$\Delta Y = (\Delta X)(\text{metal/D}_2\text{O ratio}) \quad (5)$$

The average bulk temperature rise, (ΔT_b) , was calculated for an average power density of 1 MW/liter. To obtain the bulk temperature rise in the hot channel, it is necessary to multiply the average ΔT_b by the radial ϕ_{max}/ϕ_{ave} power ratio.

The pressure drop across the core was calculated from:

$$\Delta P = \frac{(K_c - K_e)V^2 \rho}{2g_c} + \frac{0.0065}{2g_c} \left(\frac{V}{D_e} \right)^{1.865} \rho^{0.865} \mu^{0.175} L \quad (6)$$

where ΔP is the pressure drop across the core, K_c and K_e are the entrance and exit loss coefficients, D_e is the equivalent diameter of the channel, μ is the viscosity, and g_c is the gravitational constant. The above equation includes a dimensional relationship for the friction factor based on experimental data of Moody as reported in the Crane Handbook:

$$f = \frac{0.0065}{D_e^{1.73}} (N_{Re})^{-0.135}$$

The pressure drop was found to be essentially independent of metal/D₂O ratio (except for a small effect due to differences in K_c and K_e).

The burnout conditions were calculated from the following equations obtained from DPW-54-526¹:

$$(q/A)_{Bo} = h_{Bo} (t_w - t_b)_{Bo}$$

¹DPW-54-526, "Prediction of Heat Transfer Burnout", L. Bernath, August 20, 1954.

$$h_{Bo} = 5710 \left(\frac{D_e}{D_1} \right)^{0.6} + 48 \frac{V}{D_e^{0.6}}$$

$$(t_w) = 57 \ln P - 54 \left(\frac{P}{P + 15} \right) - \frac{V}{4(D_e/D_1)^{0.6}}$$

where q/A is the burnout heat flux in Pcu units, temperature is in degrees centigrade, D_e is the equivalent diameter in feet, D_1 is the wetted surface diameter in feet, V is velocity in ft/sec, and P is pressure in psia. It is claimed that these equations will predict burnout fluxes with large subcooling values (in the design range for AETR) to $\pm 15\%$.

A4.2 Flow Across Rods

The three equations for cross flow over rods corresponding to the first three equations for flat plates are:

$$Q_{max} = \frac{t_w - t_{in}}{\frac{(1.48)(0.866)p^2}{\pi h D} \sin \frac{\pi Z}{\bar{H}} + \frac{1.15p\bar{H}}{\pi V \rho C_p (p-D)} \left[\cos \frac{\pi(\bar{H}-L)}{2\bar{H}} - \cos \frac{\pi Z}{\bar{H}} \right]} \quad (1')$$

$$Z = \frac{\bar{H}}{\pi} \tan^{-1} \left(- \frac{(1.22)(0.866)(p)(p-D) V \rho C_p}{h D \bar{H}} \right) \quad (2')$$

$$(\Delta t_b)_{ave} = \frac{Q_{ave} L_p}{V \rho C_p (p-D)} \quad (3')$$

where p is the pitch (equilateral triangle arrangement) and D is the rod diameter. All other symbols are the same as for flat plates. The coefficients 1.48 and 1.15 are the hot channel factors for the film and bulk temperatures, respectively.

The heat transfer coefficients were calculated from:

$$\frac{h_m D}{k_f} = b(N_{Re})_f^n (N_{Pr})_f^{1/3} \quad (4')$$

where b and n are obtained from data by Grimson (see McAdams, "Heat Transmission", 3rd Edition, p 273). Again, physical properties were evaluated for an average film temperature of 300°F and light water properties were used except for densities.

For each tube diameter, D ; and metal/ D_2O ratio, a unique value for the pitch of rods arranged on equilateral triangles was obtained from:

$$p = \left[\frac{\pi D^2 (1 + \text{metal}/D_2O)}{3.464 (\text{metal}/D_2O)} \right]^{0.5} \quad (5')$$

The methods for calculating Z , $Q_{\max}$, and $(\Delta t_p)_{av}$ for flow across rods were essentially the same as for flat plates.

The pressure drop for flow across tubes was calculated from:

$$\Delta P = 2 \left(0.23 + \frac{0.11}{\left(\frac{p}{L} - 1 \right)^{1.08}} \right) \frac{NV^{1.85} \rho^{0.85} \mu_f^{0.15}}{g_c D^{0.15}} \quad (6')$$

where N is the number of rod rows. This equation applies only for p/D ratios in the range of 1.5 to 4 and predicts high values for p/D ratios below 1.5. For $p/D = 1.25$ the predicted ΔP from the above equation is nearly 50% high. Accordingly, pressure drop calculations were made only for metal/ D_2O ratio of 0.7 ($p/D = 1.42$). However, for a $p/D = 1.25$, 50% of the calculated ΔP was a little lower than that calculated for a p/D of 1.42. It was therefore assumed that the pressure drop values were the same for all metal/ D_2O ratios used in this report.

A4.3 Hot Channel Factors

Although it is felt that local boiling is not a serious problem and that the reactor could be operated with some boiling (at least with wall temperatures at saturation temperatures -- the MTR now operates with wall temperatures close to the water saturation temperature), for this study reasonable hot channel factors were taken into account. The following factors were considered:

1. Errors in power distribution calculations

It is assumed that the error in power distribution calculations is 10% and therefore $F_{\text{bulk}} = 1.10$ and $F_{\text{film}} = 1.10$.

2. Errors in power measurement

It is assumed that the error in power measurement is about 5% and therefore $F_{\text{bulk}} = 1.05$ and $F_{\text{film}} = 1.05$.

3. Accuracy of heat transfer equations

It is assumed that the heat transfer equations used will correlate the data for flat plates to $\pm 10\%$ and for flow across rods to $\pm 20\%$. The F_{film} factors are:

Flat Plates

$$F_{\text{film}} = 1.10$$

Rods

$$F_{\text{film}} = 1.20$$

4. Fuel concentration tolerance

It is assumed that the fuel concentration can be maintained to $\pm 1\%$ at any one spot and that the factors are $F_{\text{bulk}} = 1.005$ and $F_{\text{film}} = 1.01$.

5. Fuel core dimensions

It is assumed that the fuel core dimensions can be maintained to ± 0.001 inches regardless of size of fuel element.

The hot channel factors then depend on fuel element dimensions, and are:

Flat Plates

$$F_{\text{bulk}} = \frac{1 + 0.001}{2 X_c}$$

$$F_{\text{film}} = \frac{X_c + 0.001}{X_c}$$

where X_c is the nominal core thickness.

Rods

$$F_{\text{bulk}} = 1.00 \text{ (mixing between channels)}$$

$$F_{\text{film}} = \frac{(D_c + 0.001)^2}{(D_c)^2}$$

where D_c is the nominal diameter of the core.

For the present study the following "average" factors were used:

Flat Plates

$$F_{\text{bulk}} = 1.02$$

$$F_{\text{film}} = 1.05$$

Rods

$$F_{\text{bulk}} = 1.00$$

$$F_{\text{film}} = 1.10$$

6. Fabrication tolerances

It is assumed that the channel widths can be maintained to ± 0.004 inches. This will affect both the flow and heat transfer coefficient. Compensating effects due to heat transfer to adjoining channels must also be taken into account (as temperatures in one channel increase more heat will be transferred to adjoining channels). The factors are particularly difficult to estimate for rods since each rod is bounded by four channels.

Flat Plates

$$F_{\text{bulk}} = 0.5 \left(\frac{\Delta X_2}{\Delta X_1} \right) + 0.5 = 1.10 \text{ (assumed the average)}$$

$$F_{\text{film}} = 0.5 \left(\frac{\Delta X_2}{\Delta X_1} \right)^{0.6} + 0.5 = 1.04 \text{ (assumed the average)}$$

Rods

$$F_{\text{bulk}} = 1.00$$

$$F_{\text{film}} = 1.02 \text{ (assumed the average)}$$

The factors are summarized in the following table:

Table A4.3 Summary of Hot Channel Factors

<u>Uncertainty Factor</u>	<u>F_{bulk}</u>	<u>F_{film}</u>
1. Power distribution calculations	1.10	1.10
2. Power measurement	1.05	1.05
3. Heat transfer coefficient calculations		
a. Flat plates	1.00	1.10
b. Rods	1.00	1.20
4. Fuel concentration tolerance		
a. Flat plates	1.005	1.01
b. Rods	1.00	1.01
5. Fuel core dimension tolerances		
a. Flat plates	1.02	1.05
b. Rods	1.00	1.10
6. Channel width tolerance		
a. Flat plates	1.10	1.04
b. Rods	1.00	1.02
TOTALS		
a. Flat plates	1.275	1.350
b. Rods	1.150	1.480

The first term in the denominator of equations (1) and (1') is related to the film temperature and the second term to the bulk water temperature. The hot channel factors can then be applied directly as multiplication factors to the appropriate terms. In addition, the fabrication tolerances affect the point of maximum wall temperature. These factors can be applied to the tangent⁻¹ terms in equations (2) and (2').

GENERAL REACTOR ARRANGEMENT PLAN VIEW		
DRAWN: MET	DRAWING NO.	REV.
SCALE: AS NOTED	FIGURE 2.A

FIGURE 2.D - RING TYPE CURVED PLATE
FUEL ASSEMBLY FOR ANNULAR CORE. FIG. 2.D

FIGURE 2.E BOX-TYPE FLAT-PLATE FUEL ASSEMBLY FOR ANNULAR REACTOR CORE.

FIG.2.E

FIG. 2.F SCHEMATIC OF BORATED D₂O REFLECTOR CONTROL SYSTEM.

FIGURE 2.F

FIG. 2.G SCHEMATIC OF VARIABLE REFLECTOR
HEIGHT CONTROL SYSTEM.

FIGURE 2.G

Fig. 2.H Thermal, Intermediate, and Fast Neutron Fluxes at Midplane in Typical Gas-Cooled Loop Test Reactor - Without Test Section

Fig. 2.I Thermal Neutron Flux at Midplane in a Typical Gas-Cooled Loop Test Reactor - With and Without Test Section

Fig. 2.J Thermal, Intermediate, and Fast Neutron Fluxes at Midplane
in Typical Water-Cooled Loop Test Reactor - Without Test Section

Fig. 2.K Thermal, Intermediate, and Fast Neutron Fluxes at Midplane
in Typical Water-Cooled Loop Test Reactor - With Test Section

Fig. 2.L Thermal Neutron Flux at Midplane in Typical Water-Cooled Loop Test Reactor

FIGURE 3.A. Thermal Neutron Flux at Midplane in D₂O Cooled Reactor with H₂O Test Loop in D₂O Island - 15 cm. Thick Core

FIGURE 3.B. Thermal Neutron Flux at Midplane in D₂O Cooled Reactor
with H₂O Test Loop in D₂O Island - 8 cm. Thick Core

FIGURE 3.C. Thermal Neutron Flux at Midplane in D₂O Cooled
Reactor with 40 cm. Radius D₂O Island-15 cm Thick Core.

FIGURE 3.D. Thermal Neutron Flux at Midplane in D₂O Cooled Reactor with 4 cm H₂O between D₂O Island and Core

FIGURE 3.E. Reactor Power vs Annulus Thickness for H₂O Annuli
Around H₂O Cooled Test Loop in D₂O Cooled Cores

FIGURE 3.F. Reactor Power vs Annulus Thickness for H₂O Annuli
Around H₂O Cooled Test Section in D₂O Cooled Cores

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FIGURE 3.G. Reactor Power vs Annulus Thickness for H₂O Annuli
Around H₂O Cooled Test Section in D₂O Cooled Cores

FIGURE 3.H. Maximum Power Density vs Annulus Thickness for H₂O Annuli Around H₂O Cooled Test Section in D₂O Cooled Cores

FIGURE 3.I. Thermal Neutron Flux at Midplane in Reactor with 2cm.
H₂O Annulus around Test Hole - 3.5 cm. Thick Core

FIGURE 3.J. Thermal Neutron Flux at Midplane in Reactor with 4 cm.
H₂O Annulus around Test Hole - 3.5 cm. Thick Core

FIGURE 3.K. Thermal Neutron Flux at Midplane in Reactor with 6 cm.
H₂O Annulus around Test Hole - 3.5 cm. Thick Core

FIGURE 3.1. Thermal Neutron Flux at Midplane in Reactor with 8 cm.
H₂O Annulus around Test Hole - 3.5 cm. Thick Core

FIGURE 3.M. Thermal Neutron Flux at Midplane in Reactor with 4 cm.
H₂O Annulus around Test Hole - 6 cm. Thick Core

FIGURE 3.N. Thermal Neutron Flux at Midplane in Reactor with 4 cm.
H₂O Annulus around Test Hole - 10 cm. Thick Core

FIGURE 3.0. Thermal Neutron Flux at Midplane in Reactor with 4 cm.
H₂O Annulus around Test Hole - 16 cm. Thick Core

FIGURE 3.P. Reactor Power vs Annulus Thickness for Various Annuli Between H₂O Cooled Test Assembly and D₂O Cooled Core

Fig. 3.Q Reactor Power vs M/W Ratio for 4 cm. H₂O Annulus
around H₂O Cooled Test Loop in D₂O Cooled Cores

Fig. 3.R Reactor Power vs M/W Ratio for 4 cm. H₂O Annulus Around
H₂O Cooled Test Section in D₂O and H₂O Cooled Cores

Figure 3.S. Maximum Reactor Power Density vs M/W Ratio in H_2O and D_2O Cooled Reactor Cores

FIGURE 3.T. Neutron Multiplication Constant vs M/W Ratio for Reactor with 4 cm H₂O Annulus around Test Hole-3.5 cm Thick Core

Fig. 3.U Required Reactor Power vs Reflector Thickness for Typical Water-Cooled Loop Test Reactor

Fig. 3.V Neutron Multiplication Constant vs Reflector Thickness for
Typical H₂O Test Loop Reactor - H₂O Cooled Reactor Core

Figure 4.A. Maximum Permissible Power Density vs Channel Width,
Flat Plates-D₂O at 30 ft/sec

FIGURE 4.B. Maximum Permissible Power Density vs Channel Width Flat Plates - D₂O at 20 ft/sec

Figure 4.C. Reactor Pressure Drop vs Channel Width Flat Plates-D₂O

FIGURE 4.D. Average Bulk D_2O Temperature Rise for 1 MW/liter Average Power Density vs Metal/ D_2O Ratio-Flat Plates

FIGURE 4.E. Maximum Power Density vs Rod Diameter for Flow Across Rods - D_2O at 15 ft/sec

FIGURE 4.F. Maximum Power Density vs Rod Diameter for Flow
Across Rods - D₂O at 10 ft/sec

FIGURE 4.G. Reactor Pressure Drop vs Rod Diameter for
D₂O Flow Across Rods

FIGURE 4.H. Average Bulk D_2O Temperature Rise for 1 MW/liter
Across Power Density vs Metal/ D_2O Ratio-Cross Flow

FIGURE A3.A

FAST GROUP DIFFUSION COEFFICIENT IN $\text{Al-D}_2\text{O}$
10 kev to 10 mev

FIGURE A3.B FAST GROUP SLOWING DOWN CROSS SECTION IN Al-D₂O
10 kev to 10 mev

FIGURE A3.C INTERMEDIATE GROUP DIFFUSION COEFFICIENT IN Al-D₂O
THERMAL TO 10 kev

FIGURE A3.D INTERMEDIATE GROUP SLOWING DOWN CROSS SECTION IN $\text{Al-D}_2\text{O}$
THERMAL TO 10 kev

FIGURE A3.E THERMAL DIFFUSION COEFFICIENT IN Al-D₂O
T = 170°F

FIGURE A3.F AVERAGE FISSION CROSS SECTION OF U(25)

10 kev to 10 mev

FIGURE A3.G AVERAGE FISSION CROSS SECTION OF U(25)
THERMAL TO 10 kev

FIGURE A3.H AVERAGE THERMAL FISSION CROSS SECTION IN U(25)
T = 170°F

FIGURE A3.I AVERAGE THERMAL ABSORPTION CROSS SECTION IN U(25)
T = 170°F

FIGURE A3.J FAST GROUP AGE IN Al-H₂O
10 kev - 10 mev

FIGURE A3.K FAST GROUP DIFFUSION COEFFICIENT IN Al-H₂O
10 kev - 10 mev

FIGURE A3.1 FAST GROUP SLOWING DOWN CROSS SECTION IN Al-H₂O
10 kev - 10 mev

FIGURE A3.M INTERMEDIATE GROUP AGE IN Al-H₂O
THERMAL - 10 mev

FIGURE A3.N INTERMEDIATE GROUP DIFFUSION COEFFICIENT IN Al-H₂O
THERMAL - 10 kev

FIGURE A3.0 INTERMEDIATE GROUP SLOWING DOWN CROSS SECTION IN Al-H₂O
THERMAL - 10 kev

FIGURE A3.P THERMAL DIFFUSION COEFFICIENT IN Al-H₂O
T = 170°F

FIGURE A3.Q FAST GROUP AGE IN Zr-H₂O
10 kev - 10 mev

FIGURE A3.R FAST GROUP DIFFUSION COEFFICIENT IN Zr-H₂O
10 kev - 10 mev

FIGURE A3.8 FAST GROUP SLOWING DOWN CROSS SECTION IN Zr-H₂O
10 kev - 10 mev

FIGURE A3.T INTERMEDIATE GROUP AGE IN Zr-H₂O
THERMAL - 10 kev

FIGURE A3.U INTERMEDIATE GROUP DIFFUSION COEFFICIENT IN Zr-H₂O
THERMAL - 10 kev

FIGURE A3.V INTERMEDIATE GROUP SLOWING DOWN CROSS SECTION IN Zr-H₂O
THERMAL - 10 kev

FIGURE A3.W THERMAL DIFFUSION COEFFICIENT IN Zr-H₂O
T = 170°F