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VARIABLE MODERATOR REACTOR

DEVELOPMENT PROGRAM

ATL Job 11003

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FOREWORD

Development of the Variable Moderator Reactor (VMR) is conducted under Contract No. AT(04-3)-250, Project Agreement No. 3, by Advanced Technology Laboratories, a Division of American-Standard, for the U.S. Atomic Energy Commission. This second quarterly technical progress report covers the continuation of work under Phase I of the contract, "Physics and Kinetics Analysis and Initial Evaluation," and the initiation of work under Phase II, "Critical Experiment and Analysis of Results."

The information contained in this progress report is preliminary and subject to change as the program evolves.

PROGRAM OBJECTIVES

The objectives of the work currently being performed, and of the work to be performed during the next quarter (Nov. 16, 1959 through May 15, 1960), in accordance with the agreed upon scope,¹ is as follows:

1. To complete the development and testing of all portions of the boiling core physics model PUREE',² and to modify the code as dictated by future checking against experimental data.
2. To use the hydrodynamic code BOCH³ and the physics codes PUREE' and VAL PROD⁴ to determine the relationship between voids, flux, and burnup, and to investigate the relationship between $k_{\text{effective}}$ and void fraction, including the effects of the spatial distribution of steam voids, and neutron flux.
3. To complete the analog representation of the VMR core kinetics, and to use this representation, together with results from "2" above, to study the transient characteristics of various VMR core arrangements. To compare the results of the analog representation with existing reactor transient response data.
4. To design the critical experiment facility, and to procure and fabricate equipment for the facility as needed.
5. To plan the experimental program in detail, to prepare the hazards report, and to commence the experiments.

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1. "Program for Evaluation of the Variable Moderator Reactor under the new Reactor Concepts Program for the Atomic Energy Commission", Atomic Energy Division of American-Standard, Proposal P-470A, December 17, 1958.
 2. PUREE' is a non-dimensional physics code which computes homogenized constants for use in neutron-balance equations.
 3. BOCH is a two-dimensional hydrodynamic code which computes the non-uniform void distribution in the core.
 4. R. H. Stark and C. H. White, "VAL PROD, a Revision of the PROD II 630 Multigroup," General Electric, Atomic Power Equipment Division, GEAP-0952, January 9, 1957 (a one-dimensional physics code for solving criticality and reactivity).

STATUS OF PROGRAM

The following sections describe the progress in each of the above areas and in the areas outlined under "Objectives of Current Work" from the previous Quarterly Progress Report.⁵

A. Hydrodynamics

The hydrodynamics code BOCH has been employed to obtain the relationship between a large number of variables (e.g. voids, flux, pin spacing, power density, pressure) for VMR lattices. Based upon these relationships, a reference core design has been selected for examination, using the integrated physics and hydrodynamics analytical models.

B. Kinetics

The relationships describing the individual blocks of the kinetics model block diagram have been completed.

C. Physics

1. "Simplified" Analysis

A "simplified" analysis has been completed, and the design parameters have been established. A check of these parameters, using PUREE', has been initiated.

2. PUREE'

The first three energy groups and the thermal group have been checked against experimental lattices. Portions of the developed code are being used to check the critical experiment design and to investigate the "reference" core design.

D. Critical Experiment

1. Fuel

A purchase order has been awarded to Mallinckrodt Nuclear Corp. for the conversion of UF_6 to UO_2 and for the fabrication of the fuel rods.

2. Critical Experiment Design and Equipment Procurement

The critical experiment facility design is approximately 60% complete. Quotations have been requested for some of the equipment, and fabrication of the fuel-pin grid plates has commenced.

5. "Variable Moderator Reactor Development Program," Advanced Technology Laboratories, a Division of American-Standard, Quarterly Progress Report No. 1, ATL-A-100, 31 August 1959.

3. Experimental Program

A complete preliminary program has been planned. This planning includes a consideration of the number, kinds, and schedule of experiments to be performed.

4. Hazards Report

Portions of the PUREE' physics analysis are being used to generate information for the hazards report. Battelle Memorial Institute is preparing the report, consistent with the planned program.

SUMMARY OF RESULTS

A. Reference Core Design and Parametric Studies

Based upon early physics results from the "simplified" analysis, from the BOCH hydrodynamic parameter studies completed to date, and from work performed by ATL prior to the award of this contract, a reference core design has been selected for study. This reference core represents the first in a series of such reference designs, rather than, an optimized core. Optimization will occur as a result of many reference design and parametric studies. The purpose of selecting a reference design is to provide a model for performing integrated hydrodynamics-physics analyses, and to perform certain parametric studies. The design parameters of this first reference core are shown in Table II of Section I, "Hydrodynamics." The reasons for selection of various of these parameters are summarized below.

Because of the separation of the moderator from the coolant in the VMR, it is desirable to arrange the fuel rods in a triangular pitch and in hexagonally shaped bundles in order to obtain appropriate ratios of water to uranium oxide. Therefore, a triangular-pitched fuel array in hexagonal shroud assemblies has been selected for study in the reference design. Other arrays over a range of water-to-uranium-oxide ratios will, of course, be examined, including fuel bundles in circular shrouds.

The selection of 61 pins per fuel assembly listed in the reference design is based upon early physics results. In the critical assembly studies, 91 pins per assembly (for the next larger hexagonal array) resulted in a less reactive core than the 61-pin assemblies for the same total quantity of fuel (2500 kg of UO_2). Another calculation for 37-pin assemblies and

with a total of 2500 kg of UO_2 resulted in the most reactive core studied. In order to arrange the same total quantity of fuel in this latter core as with the 61-pin assembly, 40% more fuel assemblies were required, which in turn resulted in 40% more fuel shrouds. The expense associated with the fabrication of these shrouds appears to offset any advantage accruing from the greater reactivity of the 37-pin assemblies. Therefore, 61-pin assembly fuel elements were selected for the reference case.

The 70 mw of thermal output was selected for the reference design because it represents approximately the power needed to operate a 20-mw turbine-generator plant, and because there is confidence that a natural-circulation reactor can be designed for this rating.

The value of 0.07 used for the exit quality of the water (and the exit steam voids) is based upon values used in other operating reactors and reactor designs, such as the Experimental Boiling Water Reactor (EBWR), at Argonne National Laboratory. The value is conservative with respect to the transient stability of these other reactors.

The pin spacing to-diameter ratio (S/D) in the reference design is based upon both hydrodynamics and physics considerations. In the studies completed for the critical experiments, the S/D was varied over the range 1.2 to 1.45. Spacings within this range appear to impart desirable reactivity-void characteristics to the core. Hydrodynamics considerations favor spacings toward the upper end of this range, since considerably larger power densities can be obtained than with smaller spacings. Therefore, based upon a core diameter selected from previous ATL work, and on the thermal output and the exit quality indicated above, an S/D capable of satisfying all these requirements was selected. This S/D was 1.39.

The fuel-pin diameter, the fuel-pin-cladding thickness and material, the shroud thickness and material, the active core height, and the moderator gap between shrouds are based upon preliminary design work performed by ATL prior to the award of this contract.

The radial and axial peak-to-average-flux ratios are based upon Bessel and sine functions respectively. The Bessel function for the radial flux distribution is a J_0 function with 7-cm reflector savings on the radius. These assumed distributions will be adjusted by computing the actual distributions with PUREE'.

All of the parameters discussed above, and others, will be varied around the reference design in order to obtain optimum arrangements. Some of these parametric studies have been completed and are shown graphically in Section I, "Hydrodynamics."

B. Critical Experiment Design and Testing Procedures

The basic requirements of the critical facility have been established. The facility is arranged so that various core arrangements will be examined over a range of variables (see Table II of Section IV, "Critical Experiment"). The variables selected represent those core arrangements which fall within the range of actual reactor arrangements to be investigated by PUREE'. Since it is believed that PUREE' can be made to closely simulate many VMR core arrangements, it is of primary importance to check it against experimental lattices of the type encountered in the VMR.

I. HYDRODYNAMICS

A. Parametric Studies

The BOCH code, programmed for the IBM 650 computer, has been applied to the calculation of the hydrodynamic aspects of the performance of various fuel elements. Curves typical of the results of the BOCH calculations are shown in Figure 1. The graph presents, as a function of heat output per pin, the following:

- 1) Mass flux, G .
- 2) Static pressure difference, ΔP_c , between the plenum below the element and the bottom of the convection riser extending above the elements.
- 3) The length-average density, $\bar{\rho}$, of the water in the element.
- 4) The density, ρ_e , of the two-phase mixture leaving the element.

The parameters held constant for these curves are indicated by the conditions listed on the graph.

Figure 1 is typical of a large family of similar curve sheets that have been prepared. Each member of the family presents the same information but differs in the parameters held constant during the calculations. Thus, by cross plotting, it is possible to show, for example, the heat output per pin as a function of pin spacing-to-diameter ratio (the number of pin diameters between adjacent pin centerlines in delta array) when the static pressure difference, ΔP_c , across the element is held constant. Figures 2, 3, and 4, discussed below, have been obtained by this method. In these graphs, the static pressure difference, ΔP_c , is one of the constant parameters and is specified to be 400 psf (2.78 psi). This value is selected on the basis that it is in the pressure drop range suitable for a reactor in which the water is recirculated by natural convection.

Figure 2 shows the heat output per pin for an element operating at reactor pressures ranging from 400 to 1400 psi. It will be noted that the pressure drop ΔP_c , the inlet subcooling of the water, and the fuel-element geometry (specified by pin diameter, pin spacing, and pin length) are constant parameters. In preparation of this graph, an attempt was made to select the exit quality of the coolant in such a manner that the length-average density, $\bar{\rho}$, would be constant with pressure. Although it is apparent that this objective was not entirely realized, the graph does indicate the trend of attainable heat output per pin as a function of

TYPICAL CALCULATION RESULTS OBTAINED WITH USE OF BOCH CODE

FIGURE 1

EFFECT OF PRESSURE ON HEAT OUTPUT PER PIN WHEN
 PRESSURE DIFFERENCE ACROSS ELEMENT IS 400 PSF

FIGURE 2

EFFECT OF PRESSURE ON RELATION BETWEEN
 LENGTH-AVERAGE DENSITY AND HEAT OUTPUT WHEN
 PRESSURE DIFFERENCE ACROSS ELEMENT IS 400 PSF

FIGURE 3

EFFECT OF PIN SPACING/DIAMETER ON DENSITY AND HEAT OUTPUT PER PIN
WHEN PRESSURE DIFFERENCE ACROSS CORE IS 400 PSF

FIGURE 4

pressure. It will be observed, for example, that operation of an element at 1000 psia will permit almost twice the heat output attainable at a pressure of 400 psia. At pressures above 800 psia, the curve indicates that the rate of increase in heat output diminishes rather rapidly with increasing pressure. Figure 3 is presented to demonstrate the thermal performance of an element operating at pressures above 800 psia.

Inspection of Figure 3 reveals that for the conditions specified, the maximum heat output at any selected length-average density ($\bar{\rho}$) occurs at a pressure of 1000 psia. The information presented on the graph does not, of course, represent all the considerations applicable to the hydrodynamic design of a reactor core and thus it cannot be concluded, on the basis of Figure 3, that 1000 psia is an optimum pressure from a hydrodynamics viewpoint. Nevertheless, on the basis of this graph (and the known trend in the BWR field), a pressure of 1000 psia has been selected for a VMR reference design.

Figure 4 shows the heat output per pin and the length-average and exit densities as functions of pin spacing-to-diameter ratio and exit quality of the two-phase mixture. It will be noted that the conditions represented on the graph correspond to a pressure level of 1000 psia and a static pressure difference of 400 psf. The advantage of wide pin S/D and large exit quality is evident. The advantage of wide pin S/D is also shown, in terms of attainable power density, on Figure 5, which is based on the 0.07 exit quality curve of Figure 4.

Figures 1 through 5 are based on a sine distribution of heat input:

$$\frac{Q/L(Z)}{(Q/L)_{avg}} = \frac{\pi}{2} \sin \left(\frac{\pi Z}{L} \right)$$

In the actual reactor, the axial flux will not be a sine function, since the axial neutron flux distribution (and thus the axial heat output distribution) is dependent on the axial distribution of density. For this reason, it is of interest to examine the influence of the shape of the axial heat input distribution on the performance of a fuel element. Figure 6 shows three axial distributions of heat output and the corresponding axial distributions of coolant density. The heat output distribution curve, labeled S, is a sine curve. The other two, mirror images of each other, are arbitrarily skewed. The total heat output rate is the same for all three curves. Perhaps the most interesting conclusion to be drawn from Figure 6 is that the static

EFFECT OF PIN SPACING/DIAMETER ON MAXIMUM POWER DENSITY
WHEN PRESSURE DIFFERENCE ACROSS ELEMENT IS 400 PSF

FIGURE 5

EFFECT OF HEAT INPUT DISTRIBUTION ON ELEMENT PERFORMANCE

FIGURE 6

pressure difference between the ends of the element for the skewed distributions differs by only $2\frac{1}{2}\%$ from the value obtained for the sine heat output distribution. This fact implies that the heat output distribution will have negligible effect on the performance of the element. The reasons leading to the small difference in ΔP_c can be drawn from an inspection of the five pressure-difference components that compose the static pressure difference, ΔP_c . These components for the three output distributions are given in Table I.

TABLE I
Components of the Static Pressure
Differences Presented in Figure 6

Static Pressure Difference (psf) due to	Distribution Curve		
	L	S	R
1. Friction in element	186.2	164.1	142.8
2. Acceleration of flow	30.4	30.4	30.4
3. Exit from fuel element	- 17.3	- 17.3	- 17.3
4. Entrance to fuel element	26.9	26.9	26.9
5. Hydrostatic head	178.3	189.9	201.2
Total, ΔP_c	404.5	394.1	384.1
Difference relative to S	+ 10.4	ref.	- 10.0

Inspection of Table I reveals that the increase in the friction pressure difference with curve L (relative to curve S) is partially compensated by the decreased hydrostatic head. The converse effects are observed with curve R.

Presuming that this conclusion is born out by further studies leads to the further conclusion that the recirculation rate, and consequently the inlet subcooling, for a reactor core will be largely independent of the axial flux distribution. In this event, the calculations will be simplified, since the entire BOCH program will not be involved in reiterations between the physics codes and the BOCH program when finding the axial density distribution consistent with the requirements of both physics and hydrodynamics.

B. Reference Design

A reference core design, intended to provide a model to which the physics and kinetics analyses can be applied, has been derived on the basis of the foregoing material. It must be emphasized that the reference design does not represent an optimized design and alterations in the design due to physics, kinetics, and economic considerations are to be expected. The reference design is summarized in Table II. Fuel assemblies in the core are presumed to be orificed in such a manner that, at design conditions, the exit quality is the same for all assemblies. To an accuracy consistent with the purposes of this reference design, the axial density distribution can therefore be considered to be the same for all assemblies. The axial distribution of density based on central fuel assembly conditions is shown in Figure 7.

TABLE II
REFERENCE CORE DESIGN NO. 1

Thermal output	70 mw
Pressure	1000 psia
Pin diameter (clad OD)	0.485 in.
Pellet diameter	0.420 in.
Clad thickness	0.030 in.
Clad	Zircaloy
Pinspacing/diameter (delta array)	1.393
Pins per element	61
Elements per core	121
Shroud thickness (stainless)	0.05 in.
Distance between flats of shroud (internal)	5.35 in.
Moderator gap between shrouds	0.75 in.
Volume fractions	
Coolant volume fraction	0.4055
Moderator volume fraction	0.2275
Fuel-pin volume fraction	0.3385
Shroud volume fraction	<u>0.0285</u>
Total	1.000
Active length of fuel pin	6 ft
Exit quality of water	0.07
Exit void fraction	50%
Exit density of water	24 lb/ft ³
Length-average density of water	35 lb/ft ³
Maximum heat flux*	134,000 Btu/hr-ft ²
Radial peak to average flux	2.0
Axial peak to average flux	1.57
Feedwater temperature	459°F
Water subcooling of entrance to element	6°F
Feedwater subcooling	85.6°F
Riser height	6 to 10 ft

* Without "Engineering" hot spot factors.

AXIAL DISTRIBUTION OF DENSITY IN CENTRAL ELEMENT OF
REFERENCE DESIGN NO. 1

FIGURE 7

II. KINETICS

Formulation of a kinetics model has been completed, and the report describing it is in rough draft form. The description will be in a subsequent progress report to permit simultaneous presentation of the model and the results of analog computation performed on it.

A kinetics block diagram representing the interrelationships of the variables determining system stability of the model is shown in Figure 8. Time dependence of the variables is not indicated in the diagram but is to be understood.

LEGEND

D steam demand rate
 m mass rate of coolant through core
 P system pressure
 Q_c heat rate to coolant
 Q_g heat rate generated in fuel
 S steam generation rate
 T_{sat} saturation temperature
 T_f fuel temperature
 T_{in} temperature of coolant entering element

$\delta\rho_P$ density change due to change in P
 $\delta\rho_{T_{in}}$ density change due to change in T_{in}
 $\delta\rho_m$ density change due to change in m
 $\delta\rho_Q$ density change due to change in Q_c
 δk_{level} reactivity change due to change in moderator level
 δk_ρ reactivity change due to change in coolant density
 δk_T reactivity change due to change in fuel temperature
 δk net change in reactivity

KINETICS BLOCK DIAGRAM

FIGURE 8

III. PHYSICS, PUREE' ANALYSIS

A. Groups 1, 2, and 3 of the Four-Group Model

The cross section averaging for U 235, U 238, H, O, Al, Fe, and C was completed for the 19-group structure that forms the basis of the four-group model.⁵ This completes the preparation of data necessary for testing of PUREE', as well as computation of the Battelle experiments on the VMR lattices.

An initial test of the accuracy of this data, and the PUREE' procedure used to determine constants for the first three (high-energy) groups of the four-group model, was made by calculating the reactivity of one of the slightly enriched uranium metal lattices assembled by BNL and WAPD.⁶ This test case had a water-to-metal ratio of 3, an enrichment of 1.3%, and a fuel-rod diameter of 0.6 inch. The constants for the thermal group were based on the measured thermal utilization, f , so that the difference between the computed reactivity and unity gave a measure of the accuracy of the first three groups. The computed reactivity was too high by about 1.3%. In order to improve accuracy of the model, its component parts were compared in detail with the experiments performed by Kouts and others. These components parts are 1) ratio of U 238 fission rate to U 235 fission rate; 2) ratio of U 238 capture rate in Groups 1 + 2 + 3 to that in Group 4; 3) ratio of U 235 fission rate in Groups 1 + 2 + 3 to that in Group 4; and 4) change in buckling associated with a given addition of poison (or some other such quantity closely related to the leakage of neutrons).

Most of the discrepancy between the calculated and experimental values was due to too high a computed ratio of U 238 fission rate to U 235 fission rate. The method of making corrections for fine structure in Group 1 was altered to one of using a one-group diffusion theory solution in the cylindrical cell, with neutron removal from the group resulting from a single elastic scattering collision with hydrogen or an inelastic scattering collision with uranium. PUREE' was then rerun on the test case, and the reactivity was within 0.5% of criticality.

6. H. Kouts, et al, "Physics of Slightly Enriched, Normal Water Lattices," Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, P/1841 (1958).

The computed resonance capture agreed exactly with the experimental value quoted in Kouts' first Geneva Conference paper,⁷ although it was slightly in error (high) as compared with the value quoted in his second Geneva Conference paper.⁶ All of these component parts are within 1% in reactivity and thus appear acceptable. The small errors that do exist appear to be partially cancelling each other in the over-all reactivity result.

It was decided to test the method and constants on several more experimental lattices to determine if the accuracy would also be good for smaller rod sizes, different fuel composition, closer spacing of rods, etc. The four-group constants have thus been generated for the following additional cases: 1) water-to-uranium (metal) ratio = 2, enrichment = 1.3%, and rod diameter = 0.6 inch; 2) water-to-uranium (metal) ratio = 3, enrichment = 1.3% and rod diameter = 0.387 inch; 3) water-to-uranium dioxide ratio = 1.78, enrichment = 1.3%, and rod diameter = 0.383 inch. The VAL PROD reactivity code has not yet been run on these cases.

B. Group 4 (Thermal Group) of the Four-Group Model

A preliminary check against various experiments has been completed on the PUREE' thermal analysis, using blackness techniques, as described in the prior quarterly progress report.⁵ A further check against experiments was made of an analytic model without using blackness techniques but using methods similar to those described in the original PUREE' analysis,⁸ and as described below.

This analytic model involves the use of the P_3 ^{9,10} and POP¹¹ codes in combination with tables of Wigner and Wilkins¹² flux-weighted cross sections. The neutron-flux distribution in a cylindrical cell, consisting of one fuel pin cladding and its associated coolant, is computed by means of the P_3 code on the IBM 650. The cross sections for this calcu-

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7. H. Kouts, "Exponential Experiments with Slightly Enriched Uranium Rods in Ordinary Water," Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Vol. 5, P/600 (1955).
 8. "PUREE', A Four-Group Theory for Calculating Effective Homogeneous Cross Sections," Atomic Energy Division of American-Standard, ASAE-41, January 30, 1958.
 9. J. C. Mark, "The Spherical Harmonic Method," CRT-340.
 10. John Weil, KAPL-1173 (Confidential-Restricted Data).
 11. R. L. Crowther, "POP, A P_3 Processing Code," General Electric, Atomic Power Equipment Department, GEAP-0982, 3/15/57.
 12. E. P. Wigner and J. E. Wilkins, Jr., AECD 2275 (1948).

lation are obtained from Amster's Tables¹³ of Wigner and Wilkins flux-weighted cross sections. In these Tables, cross sections for each pertinent process are given as functions of the temperature, the ratio of the number of U 235 atoms to hydrogen atoms, the ratio of the number of Pu 239 atoms to hydrogen atoms, and the ratio of the $1/v$ absorption cross section at 0.025 volt to the number of hydrogen atoms in an equivalent homogeneous mixture of moderator and absorbers. Initially, the values of the latter three independent variables are obtained by assuming no self-shielding, that is, by uniform mixing. The resultant cross sections are used heterogeneously in the multiregion P_3 spherical harmonic approximations to obtain the distribution of the thermal flux, and for convenience the output is edited and summarized by the POP code. This code averages the cross sections weighted by the P_3 fluxes. Should the fuel-pin-flux depression obtained in these calculations be sufficient to make it necessary, the flux can be used as a weighting function to obtain new values of the three independent variables from Amster's Tables and the entire process repeated with a new set of Wigner and Wilkins cross sections. The flux obtained from this second iteration, if needed, or from the single P_3 calculations, can be used as a weighting function to obtain the effective homogeneous thermal cross sections for a fuel element. These cross sections can then be used in a final cylindrical P_3 calculation to determine the distribution of thermal neutron flux between fuel element, shroud, and moderator. The equivalent homogeneous core cross section is obtained by editing the results with the POP code and using the resultant products of relative fluxes and relative volumes as weighting functions for the three regions. A detailed thermal flux distribution may be obtained by enfolding the pin-clad-coolant and the element-shroud-moderator curves.

Because of the large number of pins per element and the large number of elements per core, the above procedure is expected to yield an excellent approximation for most of the VMR experimental lattices. For the most tightly packed lattices, however, geometric complexities are likely to cause difficulty and it may be necessary to make use of the more complex PUREE' analysis. At present, the calculation described above has been compared

13. Harvey J. Amster, "A Compendium of Thermal Neutron Cross Sections Averaged Over the Spectra of Wigner and Wilkins," WAPD-185, January 1958.

with experimental data on the Yankee Critical Experiments¹⁴ and Kouts' exponential experiments on light water slightly enriched lattices.⁷ A simplified version of this method was used in a previous American-Standard report,¹⁵ in which a comparison with a number of experiments was made. The results of these comparisons are shown in Table III. The Yankee¹⁴ and Kouts⁷ experimental check results were comparable to those listed in Table III.

A check of the more complex blackness method of computing the PUREE' thermal group against experiment¹⁶ was unsatisfactory. Efforts to find the reason for the discrepancy are continuing.

As with the first three energy groups, the thermal groups of PUREE' (using either or both of the methods described above) will be checked against other experimental lattices and against the VMR critical experiments and will be further modified as required.

Required numerical data for the VMR critical experiment and for the "reference" core design referred to in Section I, "Hydrodynamics", are being generated by the P_3 POP code method of thermal group analysis.

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14. "Yankee Critical Experiments - Measurements on Lattices of Stainless Steel Clad Slightly Enriched Uranium Dioxide Fuel Rods in Light Water," YAEC-94.
 15. Reactor Physics of H₂O-Moderated Power Reactors, ASAE-S-7, February 1, 1958.
 16. H. Kouts, et al, "Physics of Slightly Enriched, Normal Water Lattices," Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, P/1841 (1958). The case used is described in Table 6, p 461: Rod diameter = 0.388 in.; Enrichment = 1.3%; Fuel density (UO₂) = 7.52 gm/cm³; W/UO₂ Volume ratio = 1.39.

TABLE III
SUMMARY OF P₃ RESULTS FOR THERMAL UTILIZATION

Case	Rod Diameter (in.)	Enrichment (%)	Water/Metal	Material	σ_a	σ_t	$\bar{\phi}$	f (calc)	f (exp)
1	0.6	1.3	2/1	fuel	0.4883	0.882	3.62	0.884	0.890
				aluminum	0.01385	0.0964	4.216		
				water	0.0196	3.58	5.636		
2	0.6	1.3	3/1	fuel	0.4883	0.882	5.096	0.830	0.849
				aluminum	0.01384	0.0964	5.901		
				water	0.0196	3.58	8.541		
3	0.6	1.15	3/1	fuel	0.4453	0.832	5.534	0.822	0.835
				aluminum	0.01385	0.0964	6.308		
				water	0.0196	3.58	8.820		
4	0.387	1.3	3/1	fuel	0.4883	0.882	5.258	0.856	0.857
				aluminum	0.01385	0.0964	5.734		
				water	0.0196	3.58	6.933		
5	0.6	1.3	1.5/1	fuel	0.4883	0.882	9.381	0.924	0.918
				aluminum	0.01385	0.0964	10.128		
				water	0.0196	3.58	12.360		

IV: CRITICAL EXPERIMENT

A. Fuel Procurement

Purchase Order T-VC-F-1 has been released to Mallinckrodt Nuclear Corp. for the conversion of UF_6 to UO_2 , and for the fabrication of the fuel rods; 2500 kg of UO_2 enriched to 1.8% has been ordered. At least 75 of the fuel rods will be "jointed" near the bottom to permit insertion of irradiation foils. Delivery of the first lot of fuel to Battelle Memorial Institute is scheduled for January 22, 1960.

B. Critical Experiment Design and Equipment Procurement

Three preliminary drawings are included to show the critical assembly arrangement. Dwg. D-247 shows the flow schematic for the facility. As indicated on the drawing, separate and independent control of the moderator and coolant regions is effected from the control room. Furfural and furfuryl alcohol will be used in the coolant region to simulate voids.

Dwg. D-268 shows the fuel shrouds, which are arrayed vertically in the core separating the coolant region from the moderator region. Fuel rods will be spaced in different arrangements inside these shrouds. Although the drawing indicates that the shroud material is stainless steel, aluminum may be used.

Dwg. F-256 shows the arrangement of the core assembly. As noted on the drawing, the shrouds are positioned with respect to each other by the base plate, which is supported in turn by the hexagonal-shaped core-support structure of welded channels. The shrouds are bolted to the base plate from the bottom; thus, no top support is required. Different base plates will be used to accommodate different shroud spacings. Different shroud sizes will be used for fuel-rod lattices of varying pitch (and different coolant-to-uranium ratios). Three Y-shaped control rods and one single-bladed control rod enter the core between the shrouds from above. The fluid simulating coolant to the fuel region in the shrouds enters each shroud through a flexible tube, which is connected to a coolant distribution header located within the hexagonal core-support structure. The fluid simulating moderator surrounds the shrouds and is contained within the 7-foot-diameter core tank. Rapid dumping of the moderator is provided through a 4-inch dump line.

Instrumentation for the critical experiments consists of a complete nuclear and process instrument system already in existence at the Battelle facility. Inasmuch as precise measurements of the reactivity worth of the moderator and coolant are required, two very

NOTES:

1. LEGEND

SOLENOID VALVE

GLOBE VALVE

BOARD MOUNTED
MANUALLY CONTROLLED

LEVEL INDICATOR

PRELIMINARY

UNLESS OTHERWISE NOTED		REGD.		PART NO.		NAME OR DESCRIPTION		MATERIAL & MATL. SPEC.		TIME	
DIMENSIONS ARE IN INCHES		MATERIAL & OR HEAT TREAT		SIZE		NEXT ASSY		APPROVALS		CONTRACT	
TOLERANCES:								DRAWN-DATE		CHECK-DATE	
FRACTIONS		DECIMALS		ANGLES				C. PATTON		10-25-58	
REMOVE ALL BURRS.								APPRO-DATE		APPRO-DATE	
BREAK SHARP EDGES .015 X 45°								SCALE		JOB NO. 11003	
MACHINE FILLETS .01 R								DWG INDEX		SHEET OF 247	
MACHINE FINISH RMS								SHEET OF 247		SHEET OF 247	
DO NOT SCALE DRAWING								SHEET OF 247		SHEET OF 247	
FLOW DIAGRAM COOLANT AND MODERATOR SYSTEM V.M.R. CRITICAL EXPER.											
ADVANCED TECHNOLOGY LABORATORIES A DIVISION OF AMERICAN-Standard MOUNTAIN VIEW, CALIFORNIA											
ATL-18 REV. 4/58/59

SECTION X-X

DETAIL B
SCALE: 24" = 1'-0"

BRIDGE 1/8 x 1/8 x 1/2 S5T
TACK WELD IN
LOCATION SHOWN

DETAIL A

NOTES:

1. DEVIATION OF ACTUAL SHAPE OF CAN SHALL NOT EXCEED $\pm \frac{1}{64}$ FROM TRUE HEXAGONAL SHAPE.
2. ACTUAL ϕ OF CAN SHALL NOT DEVIATE MORE THAN $\frac{1}{64}$ FROM TRUE ϕ .
3. MAX. PERMISSABLE TWIST IN CAN $\frac{1}{2}^\circ$.
4. FABRICATION SHALL BE IN ACCORDANCE WITH ATL SPEC. RED-109.

PRELIMINARY

DET. NO	X DIM
268-1	4.571
268-2	4.944
268-3	5.490

REQD	PART NO.	DATE	NAME OR DESCRIPTION	MATERIAL & MATL. SPEC.	FILE NO.
UNLESS OTHERWISE NOTED	PER ATL SPEC				
DIMENSIONS ARE IN INCHES	RED-109				
FRACTIONS: 1/32, 1/16, 1/8, 1/4, 1/2, 3/4, 1, 1 1/2, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 38, 39, 40, 41, 42, 43, 44, 45, 46, 47, 48, 49, 50, 51, 52, 53, 54, 55, 56, 57, 58, 59, 60, 61, 62, 63, 64, 65, 66, 67, 68, 69, 70, 71, 72, 73, 74, 75, 76, 77, 78, 79, 80, 81, 82, 83, 84, 85, 86, 87, 88, 89, 90, 91, 92, 93, 94, 95, 96, 97, 98, 99, 100					
REMOVE ALL BURRS.					
BREAK SHARP EDGES .015 X 45°					
MACHINE FILLETS .01 R					
MACHINE FINISH RMS					
DO NOT SCALE DRAWING					
APPROVALS	CHECK-DATE	APPRO-DATE	APPRO-DATE	APPRO-DATE	APPRO-DATE
CONTRACT NUMBER	SCALE	FULL	NO. 11003	DWG INDEX	
ASSEMBLY DETAILS			SHROUD-STEEL		
VMR CRITICAL EXPERIMENT			268		
SHEET 1 OF 1			ATL-10 REV. 4/28/59

SECTION "C-C"

DETAIL "1"
SCALE: 5" = 1'-0"

PLAN "A-A"

PLAN "B-B"

NOTES:

1. * - DENOTES MATERIAL FURNISHED BY OTHERS.
2. FOR LIST OF MATERIAL SEE DWG. NO. C-277.

PRELIMINARY

UNLESS OTHERWISE NOTED		MATERIAL FOR REACTOR	
DIMENSIONS ARE IN INCHES		TOLERANCES	
FRACTIONS - DECIMALS - ANGLES		NEXT ASBY	
3/16 - .005		S 1/4"	
REMOVE ALL BURRS		APPROVALS	
WELD SHARP EDGES .015 ± .005		DRAWN - DATE	
MACHINE FILLETS .015 ± .005		CHECK - DATE	
MACHINE FINISH		CONTRACT NUMBER	
DO NOT SCALE DRAWING		SCALE 1 1/2" = 1'-0" 100%	
		SHEET 1 OF 3	

ASSEMBLY
REACTOR CORE
61 SHROUDS AT 5.95 PITCH
VMR CRITICAL EXPER

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MOUNTAIN VIEW, CALIFORNIA

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accurate liquid-level measuring and indicating devices will be provided. One of these, presently in use at Battelle, measures level to within at least ± 0.01 inch by lowering a plumb bob in a "hunt and seek" manner to the water surface where contact is made and an electrical current flow produced. This current flow results in a reversal of the direction of the plumb bob, causing it to rise and break contact, again reversing direction, and repeating the procedure.

Nuclear instrumentation and controls, the critical experiment, structural supports and platforms, the core tank, the control-rod drives, and the equipment associated with the "moderator" region are being supplied by Battelle Memorial Institute. The fuel, shrouds, control rods, core-support structure, fuel-rod grid plates, and equipment associated with the "coolant" region are being furnished by ATL.

C. Experimental Program

The parameters which will be investigated in the experimental program and checked against the 'PUREE' physics analysis have been defined and are indicated in Tables IV and V. These parameters represent a spread around the values (based upon the "simplified" physics analysis, and being checked by 'PUREE') that will most probably be used in the operating VMR system. Once verified, 'PUREE' can of course be used to represent many different VMR systems.

Table V lists the possible core arrangements. In addition to the variables shown, the simulated voids can be varied by using water, furfural, and furfuryl alcohol as void-simulating materials, and by leaving the shrouds empty of all coolant. Not all of the combinations shown on Table IV will be used. All conditions will, of course, be reviewed for safety considerations in the hazards analysis.

The first test to be run will probably be arrangement No. 6, which represents a possible optimum condition for studying various reactivity-void relationships. For this core arrangement, the following items will be measured:

- 1) $\Delta k/k$ per inch of moderator and coolant (where applicable) at the just critical liquid height for:
 - a) Water coolant.
 - b) Furfuryl alcohol water mixture coolant.
 - c) Furfural coolant.
 - d) No coolant.

TABLE IV
CRITICAL EXPERIMENT DESIGN DATA

Fuel Rod

Fuel pellet OD (in.)	0.415
Fuel rod OD (in.)	0.485
Fuel cladding thickness (in.)	0.028
Fuel active height (in.)	46
Total quantity of UO_2 (kg)	2500
Fuel enrichment (% U 235)	1.8
Fuel cladding material	Aluminum

Core

Coolant-simulating materials	Water, furfuryl alcohol, furfural, air
Moderator-simulating materials	Water
Control-rod materials	Boral plate (aluminum and 35% boron carbide, clad with aluminum)
Safety rods	
Number	3
Type	Y, three-bladed
Size, each blade	approx. $3\frac{1}{2}$ in. $\times$ 48 in.
Shim rod	
Number	1
Type	Flat, single-bladed
Size	approx. 2 in. $\times$ 48 in.
Shrouds	
Material	Stainless steel or aluminum
Thickness	0.050 in. (0.100 in. if aluminum)
Shape	Hexagonal
Base Plates	
Material	Aluminum
Thickness	1 in.
Number	3

TABLE V
CORE ARRANGEMENT

<u>Core Arrangement Number</u>	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>	<u>7</u>	<u>8</u>	<u>9</u>	<u>10</u>
Approximate volume ratio of coolant/ UO_2	2.1	2.1	0.8	1.2	0.8	1.2	1.8	0.8	1.2	1.8
No. of fuel rods per shroud	37	37	61	61	61	61	61	61	61	61
Center-to-center fuel rod spacing (in.)	0.748	0.748	0.582	0.633	0.582	0.633	0.703	0.582	0.633	0.703
No. of shrouds in core	61	61	37	37	37	37	37	37	37	37
Shroud diameter across flats (in.)	4.571	4.571	4.571	4.944	4.571	4.944	5.490	4.571	4.944	5.490
Center-to-center shroud spacing (in.)	5.54	5.95	5.54	5.54	5.95	5.95	5.95	6.34	6.34	6.34
Approximate volume ratio of total water/ UO_2			1.75			2.25				2.75

- 2) Macroscopic flux measurements with water coolant for
 - a) Axial flux distribution along the center fuel rod and within the moderator region for two positions in the core.
 - b) Radial flux distribution within the center fuel rod of each element in a core symmetry sector.
- 3) Microscopic flux measurements (disadvantage factor) across symmetry sectors of fuel cell for
 - a) One position in core with furfural coolant.
 - b) Two positions in core with water coolant.
- 4) Resonance escape probability with water coolant for two positions in the core.
- 5) Fast fission factor with water coolant for two positions in the core.
- 6) Neutron spectral index with water coolant for two positions in the core.

The following items will probably be measured for core arrangements No. 4 and 9:

- 1) $\Delta k/k$ per inch of moderator and coolant (where applicable) at the just critical liquid heights for:
 - a) Water coolant.
 - b) Furfural coolant.
 - c) No coolant.
- 2) Macroscopic flux measurements with water coolant for
 - a) Axial flux distribution along the center fuel rod and within the moderator region for two positions in the core.
 - b) Radial flux distribution within the center fuel rod of elements in two positions within the core.

Subsequent to these measurements, the program will be reviewed to determine the most appropriate selection of parameters next to be investigated. A tentative program for the measurement of these parameters has been prepared, but is not included in this report because of its very preliminary nature.

D. Physics

As a check on the simplified physics analysis of configurations for the Battelle experiments (discussed in the last quarterly progress report), the following assemblies are in the process of computation, using the PUREE' code. The more accurate PUREE! computational

technique will permit determination of reactivity effects, critical water heights, etc.

TABLE VI

Coolant/ UO_2 ratio	0.8	1.2	1.8
Pin pitch (in.)	0.582	0.633	0.703
Shroud diameter (across flats) (in.)	4.545	4.944	5.490
No. of shrouds required	37	37	37
Moderator plus coolant to UO_2 ratio	1.75	2.25	2.75
Shroud spacing (in.)	5.54	5.95	6.34

Each of these is being computed for water, furfuryl alcohol, and furfural as coolants; thus nine cases are being run.

The thermal group constants for the four-group analysis of these cases are being computed by 1) homogenizing the entire cell contents to obtain Wigner and Wilkins average cross sections as given in Amster's Tables¹³, 2) using the P_3 transport theory code¹⁰ to obtain the fine structure of thermal flux in the cell, and 3) using the POP code¹¹ to obtain the averaged thermal group constants.

The configuration in the third column of Table VI, with furfuryl alcohol as the coolant, is believed to be the most reactive configuration and to require the lowest moderator height. In order to establish the placement of foils for flux measurement in certain fuel rods, it is important to know just what the minimum water height will be in the Battelle experiments. Therefore, this case is being treated first by the techniques described above, and with the use of the PUREE high-energy (first three groups) analysis. Input for the PUREE code in all nine cases shown in Table VI has been worked out.

Several physics problems must be solved before submission of a hazards report for the critical experiments. Among these are 1) computation of temperature effect of change of effective thermal cross sections, 2) temperature effect of expansion of coolant and moderator, 3) temperature effect of expansion of fuel pellets, 4) temperature effect of Doppler broadening of U^{238} resonances, 5) temperature effect of Doppler broadening of U^{235} resonances, 6) effect of voids in the coolant, 7) effect of flooding an initially empty shroud with water, 8) effect of flooding all shrouds with water. The solution of these problems is

in progress. One of the useful features of the PUREE' code pertaining to item 4) is that the effect of Doppler broadening can be computed simply by changing the input constant T, the fuel temperature, since the resonance capture is computed from tabulated values of the resonance integrals and these values are functions of the fuel temperature.

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