SM-2

REACTOR CORE AND VESSEL
MONTHLY REPORT

DECEMBER 15, 1959 to FEBRUARY 5, 1960

J. G. Gallagher - Project Engineer

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INTRODUCTION

This report contains the significant progress made on the SM-2 core and vessel program during the period December 15, 1959 to February 5, 1960. A formal review of the SM-2 reactor core and vessel program was held in Schenectady, N. Y., on January 5, 1960.

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PROGRESS SUMMARY

1. The effect of increasing the width of the fuel matrix upon side plate peaking was calculated. The increase in meat width from 2.54 inches to 2.65 inches reduces the side plate peaking and fixed element by 8% at room temperature. A preliminary evaluation of the effect of a one-half inch long Eu_2O_3 flux suppressor upon core reactivity and power distribution at the bottom of the core was completed. A new non-uniform burnup calculation of the SM-2 has been initiated. An interim report covering all the nuclear analysis up to December 31, 1959 is being written. (Task 10).
2. A modification of the thermal stress equation for fuel elements has been developed and is being applied to SM-2 and various test elements. An evaluation of two dimensional temperature distribution in fuel elements and pressure vessel is underway. Initial efforts to employ the ART code to treat loss of flow kinetics have been unsuccessful. (Task 2).
3. Calculations on the stationary fuel element indicate flow forces are great enough to keep the fuel element against the top doors at all times. The top doors of the core support structure have been redesigned to improve the core flow and pressure drop. Modifications to the control rod and control rod fuel element have been made to improve flow distribution and to give better centering of the element in the tube. (Task 3).
4. Additional replies have been received from three mills interested in supplying special grades of Type 347 and Type 304 SS to restricted chemistries. Twenty fully enriched fuel plates have been sent to Alco by BMI for substitution measurements in the SM-2 mockup. Analysis of four plates from this group indicated boron loss less than 10 weight percent. Some segregation of the UO_2 was evident in these fuel plates employing spherical UO_2 . The maximum burnup on the MTR capsules by February 15 is estimated to be 50% to 70%. (Task 4).
5. A brief thermal, nuclear and hydraulic survey has been completed and the fuel plate dimensions have been finalized for the SM-2 test in the WTR. The element is being designed to simulate an SM-2 control rod fuel element as far as possible. The fuel plate active core dimensions are 0.030" x 1.490" (minimum) x 16.0". It is expected that two elements will be tested employing either three or four plates in each element. (Task 4a).
6. The design adaptation of an SM-2 element for test in the SM-1 has been completed. Preparation of special steel has begun at BMI. Two elements will be fabricated; one with reference SM-2 fuel plates, and the other

with fuel plate compositions similar to that in the MTR and ETR capsule. An effort is being made to incorporate commercial oxide from at least two sources. (Task 4b).

7. An element with depleted annealed fuel plates was given a heat treatment at 800°F for 30 hours. Rippling occurred on both outer fuel plates, believed caused by radiant heating from the furnace coils. The static autoclave phase of the fuel element corrosion testing was completed. Results indicated that the TIG welded elements exhibited only slight general attack after 500 hours exposure in 635°F simulated reactor coolant.

8. A summary report of the thermal stress testing completed by January 1, 1960 has been issued. Flow testing of a modified control rod configuration has been satisfactorily completed. Design and component selection for the full scale water rig are approximately 75% complete. Instrumentation comprises the major part of the remaining work. (Task 6).

9. Design work has been initiated on fabrication of a flow divider mockup for the critical experiment. (Task 7).

10. All components of the prototype drive have been received at the laboratory and the drive mechanism is being assembled. All material required for fabrication of the test loop has been received and approximately 90% of the erection has been completed. (Task 11).

PREVIOUS PROGRESS REPORTS IN THIS SERIES

APAE Memo 150	APPR-1B Reactor Core and Vessel Progress Report, September 16, 1958 to October 25, 1958.
APAE Memo 160	APPR-1B Reactor Core and Vessel Progress Report, October 26, 1958 to December 4, 1958.
APAE Memo 166	APPR-1B Reactor Core and Vessel Progress Report, December 5, 1958 to December 31, 1958.
APAE Memo 170	APPR-1B Reactor Core and Vessel Progress Report, January 1, 1959 to January 23, 1959.
APAE Memo 173	SM-2 (Formerly APPR-1B) Reactor Core and Vessel Review Report, December 5, 1958 to February 13, 1959.
APAE Memo 185	SM-2 (Formerly APPR-1B) Reactor Core and Vessel Progress Report, February 14, 1959 to March 24, 1959.
APAE Memo 192	SM-2 (Formerly APPR-1B) Reactor Core and Vessel Progress Report, March 25, 1959 to April 24, 1959.
APAE Memo 197	SM-2 Reactor Core and Vessel Review Report, February 14, 1959 to May 27, 1959.
APAE Memo 215	SM-2 Reactor Core and Vessel Monthly Report, May 28, 1959 to July 9, 1959.
APAE Memo 223	SM-2 Reactor Core and Vessel Review Report, May 28, 1959 to August 24, 1959.
APAE Memo 229	SM-2 Reactor Core and Vessel Monthly Report, August 25, 1959 to October 23, 1959.
APAE Memo 235	SM-2 Reactor Core and Vessel Review Report, August 25, 1959 to December 14, 1959.

TECHNICAL REPORTS ISSUED IN THIS PERIOD

APAE 54	SM-2 Critical Experiments, CE-1, J. W. Noaks, et al., November 30, 1959.
APAE Memo 239	Thermal Stress Testing of SM-2 Fuel Elements Interim Technical Report, J. A. Christenson, January 30, 1960.

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A. CORE NUCLEAR DESIGN

1. Progress (Task 10.0) - P. E. Bobe

The nuclear effects of increasing the dimensions of the fuel matrix were calculated. It was found that increasing the width of the fuel matrix results in a slight decrease in reactivity; however, a significant reduction in side plate power peaking is obtained.

The effect of the europium oxide flux suppressors upon the core reactivity and axial power distribution was calculated; calculations indicate that the reference suppressor length of 1/2 inch will adequately suppress the power spike at the bottom of the core; in addition, an amount of boron, equivalent to the poisoning effect of Eu_2O_3 , was calculated so that a critical experiment may be performed.

Various tables of calculated vs. measured reactivities, as given in the last quarterly review report, were revised due to differences in the amounts of boron used in the calculations as compared to that actually used in the critical experiments.

An interim report, covering the nuclear analysis performed upon the SM-2 during September 28, 1958 to December 31, 1959, is about 70% written. This report will include a description of the analytical models employed at Alco.

A new non-uniform burnup calculation of the reference core has been initiated utilizing the CANDLE-2, IBM-704 code. It is planned to also perform the calculation by use of the NUB-1, IBM-650 code, and compare the results.

2. Core Analysis (Task 10.1) - P. E. Bobe, B. E. Fried

a. Effect of Increasing Width of Fuel Meat Matrix

In the SM-2 fuel elements, there is considerable peaking of the power across the side plates; this local peaking is reflected in high coolant flow rate requirements. For metallurgical and thermal reasons, the size of the fuel matrix was increased, as reported in the last quarterly progress report, APAE Memo 235; increasing the width of the fuel matrix not only should decrease the side plate peaking, but also decrease the weight percents of UO_2 and B-10 in the matrix.

The nuclear effects of changing the width of the fuel matrix were calculated by use of the plate-type P_3 theory, IBM-650 code. * The effect upon the side plate flux peaking is given in Table 10-1, in which the ratio of the thermal flux at the edge of the active region to the average flux in the active region, is tabulated. The effect upon the power peaking was calculated from the thermal flux peaking by assuming a flat, fast flux distribution across the cell; therefore:

$$P_{\max} / P_{\text{avg}} = \left(\frac{\phi_{\max}}{\phi_{\text{avg}}} \right) \beta + (1-\beta)$$

where β , the fraction of the total fissions which are thermal, was taken to be 0.573. This value of β was calculated for the old reference fuel matrix dimensions and was assumed to remain constant; in reality, β is a function of the core parameters. In addition, an estimate of the effect upon core reactivity was obtained by calculating the reactivity by modified two group theory; the fast parameters were calculated for elements with the old reference fuel matrix dimensions. These results are shown in Table 10-2.

The increase in average meat width from 6.4516 cm (2.54 in.) to 6.731 cm (2.65 in.), as shown in Drawing No. R9-13-1017 reduces the calculated side plate peaking from 1.231 to 1.1424. This should reduce $Q(\Delta\theta)$, given in APAE 235, ** by about 7.2%. This is a reduction in the overall calculated max. to average power from 3.86 to 3.58.

* Byrne, B. J. and R. L. Caton, "Two Dimensional P_3 Calculation for APPR-Type Fixed Fuel Elements," AP Note 96, Issued February 14, 1958.

** J. G. Gallagher, "SM-2 Reactor Core and Vessel Review Report," APAE Memo 235, December 24, 1959.

TABLE 10-1

EFFECT OF FUEL MATRIX WIDTH UPON FLUX AND POWER PEAKING

(T=510°F; P=2000 psia; No Xenon)

Width of Fuel Matrix (cm)	$\phi_{\max}/\phi_{\text{avg}}$	$P_{\max}/P_{\text{avg}}$
6.4516 [†]	1.4032	1.2310
6.7310	1.2486	1.1424
6.84403	1.2004	1.1148
6.95706	1.1497	1.0858
7.08406	1.0889	1.0509

[†] Calculated for fuel elements having old reference dimensions.

TABLE 10-2

EFFECT OF FUEL MATRIX WIDTH UPON CORE NUCLEAR PARAMETERS

(T=510°F; P=2000 psia; No Xenon)

Parameter	Units	Width of Fuel Matrix (cm.)				
		6.4516 [†]	6.7310	6.84403	6.95706	7.08406
Σa_T	cm ⁻¹	0.315808	0.327954	0.330776	0.333112	0.335074
Σs_T	cm ⁻¹	1.380611	1.374324	1.374512	1.374752	1.375114
Σ_{trT}	cm ⁻¹	1.348380	1.357603	1.360263	1.362479	1.364371
$\nu \Sigma_{FT}$	cm ⁻¹	0.464204	0.483231	0.488243	0.492435	0.496033
K_T		1.469890	1.473471	1.476054	1.478288	1.480367
D_T	cm	0.247210	0.245531	0.245051	0.244652	0.244313
τ	cm ²	46.019895				
K_F		1.273834				
ρ		0.566571				
β^2	cm ⁻²	0.006194				
$\Delta \rho$	%	-	0.15	0.25	0.33	0.41

[†] Calculated for fuel elements having old reference dimensions.

b. Effect of Eu_2O_3 Flux Suppressors

The reference fuel elements contain a europium oxide, integral flux suppressor at the bottom of each fuel plate; the present length of these suppressors is 0.5 in. (Drawing No. R9-13-1017). Calculations were performed to determine the effect of various lengths of flux suppressors upon core reactivity and axial power distribution. Axial, one-dimensional calculations were performed using the VALPROD, IBM-650 code. The constants for the europium oxide-water-stainless steel region at the bottom of the core, are tabulated in Table 10-3. These constants were calculated by homogenizing the europium oxide, water, and stainless steel contained in this region of the core; the fast constants were calculated by use of the MUFT-III code. The cross sections for Eu were obtained from APAE Memo 120* and APAE Note 97.**

TABLE 10-3

NUCLEAR PARAMETERS OF FLUX SUPPRESSOR REGION

(T=510°F P=2000 psia)

<u>Parameter</u>	<u>Units</u>	<u>Value</u>
D_T	cm	0.29112
Σ_{aT}	cm^{-1}	0.09513
D_F	cm	1.39465
τ	cm^2	40.5399
Σ_{aF}	cm^{-1}	0.008165
P		0.76266
$\xi \Sigma_s$	cm^{-1}	0.02624

* Murray, R. L., "Group Resonance Integrals and Cross Sections for Gold, Indium, and Europium," APAE Memo 120, December 27, 1957.

** Leibson, M. J., "Europium Macroscopic Cross Sections as a Function of Europium Burnout," APAE Note 97, February 21, 1958.

The effect of the flux suppressors upon the axial power distribution, for various lengths of the suppressor region, is shown in Table 10-4. The effect of the reference 1/2-inch suppressor upon the power distribution is shown in Figure 10-1. As seen in either Table 10-4 or Figure 10-1, the 1/2-inch suppressor nearly reduces the power peak at the bottom of the core to the value of the peak inside the core, just below the control rod bank position. According to the calculations, a 3/4-inch suppressor would be more than adequate. Thermally, a ratio of 1.36 would be permissible. The loss in core reactivity due to a 1/2-inch and a 3/4-inch suppressor is about -0.55% and -0.68% respectively.

The VALPROD calculations are based upon diffusion theory which becomes most inaccurate for small homogenized regions and near boundaries. The 1/2-inch suppressor length was based upon measurements made upon the SM-1 core, * in which 1/2-inch of suppressor was sufficient to suppress the flux peaking at the top of the fuel meat in the control rod elements. To verify the actual length of suppressor necessary to suppress the power, an experimental measurement will be performed, mocking up the poisoning effect due to the europium oxide by the use of boron impregnated mylar strips. Based upon the europium oxide concentration used in the SM-1, Core II suppressors, ** 0.0618 gms of B-10 per plate is necessary for a 1/2-inch suppressor in the SM-2; this corresponds to 0.007627 gms of B-10 per square centimeter of meat area per SM-2 fuel plate.

TABLE 10-4
EFFECT OF FLUX SUPPRESSORS UPON POWER PEAKING

(Rod bank at 13" from bottom of core;
T=510°F; No xenon; 0 MWYR)

Reflector Configuration at Bottom of Core	P (max. inside core)	P (bottom of core)	P _{bottom} P _{max.}
Water (no suppressor)	1.63	2.71	1.66
1/2-inch suppressor, then water	1.67	1.70	1.02
3/4-inch suppressor, then water	1.68	1.45	0.86

* MacKay, S. D., Project Engineer, "SM-1 Research and Development Program, Interim Report No. 2 on Core Measurements, Task No. VII, " APAE Memo No. 206, Issued June 30, 1959.

** Williamson, T. G., Editor, "Reactor Analysis of APPR-1, Core II, " APAE No. 32, Issued July 15, 1958.

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SM-2 AXIAL POWER DISTRIBUTIONS - 510°F - CLEAN - 0 MW/YR

FIG. 10-1

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c. Reference Loading of SM-2

In regard to the B-10 content of the SM-2 elements, distinction should be made between the actual amount fabricated into the plates and the effective amount present in a nuclear sense. The effective amount is less than the actual amount present due to self-shielding effects. The actual amount must be multiplied by the self-shielding factor to obtain the effective loading; the average self-shielding factor at 510°F for the SM-2 is 0.89. Therefore, the actual core loadings as compared to the effective loadings are tabulated in Table 10-5.

TABLE 10-5
SM-2 CORE LOADINGS

<u>Description</u>	<u>Actual Loading</u>	<u>Effective Loading</u>
U-235 per plate, gms		
Fixed element	46.30	46.30
Control rod element	40.56	40.56
B-10 per plate, gms		
Fixed Element	0.08106	0.07214
Control rod element	0.07102	0.06321
Wt. % UO_2 in matrix	24.5	-
Wt. % natural boron in matrix	0.191	-
Atom % U-235 in meat	5.2	-
Atom % B-10 in meat	0.21	-
Total U-235 in core, kg	36.2	36.2
Total B-10 in core, gms	63.4	56.42

d. SM-2 Nuclear Analysis Interim Progress Report

An interim report, covering the nuclear analysis performed on the SM-2 from September 28, 1958 to December 31, 1959 is being written.

This report will describe the analytical models employed at Alco and will present the results of most of the nuclear analysis performed upon the SM-2 core and vessel. This report is approximately 70% written; a tentative outline of the chapters contained in this report is as follows:

Introduction

- 1.0 Computational Models
- 2.0 Nuclear Survey
- 3.0 Steel vs. Water Reflector
- 4.0 Non-Uniform Burnout Calculation of Core
- 5.0 Two Pass vs. One Pass Core
- 6.0 Effect of Design Changes Upon the SM-2 Core
- 7.0 Miscellaneous Core Calculations
- 8.0 Flux and Power Distributions
- 9.0 Analysis of Critical Experiments
- 10.0 Control Rod Calculations
- 11.0 Review of Computational Models
- Appendix A - Microscopic Cross Sections
- Appendix B - Core and Element Configurations
- Appendix C - Adequacy of Core and Vessel Design

- 3. Analysis of Critical Experiments (Task 10.2) - P. E. Bobe, B. E. Fried
 - a. Revision of Reactivity Tables Presented in APAE Memo 235

The last quarterly progress report contained a fairly extensive comparison between analytical and experimental values of reactivity and between calculated values as calculated by various IBM codes. However, many of the values were calculated for cores containing slight differences in boron content as compared to what was actually loaded into the SM-2 Initial Mockup Core during the flexible critical experiments; this was due to performing the calculations before the final boron analysis of the experimental core had been made. Therefore, in order to put the comparisons on a uniform basis, the values were extrapolated to the actual B-10 content which was loaded into the core; an average B-10 worth of 0.3% ρ per gram of B-10 was used. *

The equivalence of various IBM codes in the calculation of reactivity is shown in Table 10-6. The reactivities of various configurations of the SM-2 experimental core at 68°F and for the reference core at 510°F are given in Tables 10-7 and 10-8 respectively. The best estimate of reactivity of the SM-2 reference core under various core conditions is given in Table 10-9.

* Noaks, J. W., et al, "SM-2 Critical Experiments," APAE No. 54, Issued November 30, 1959.

TABLE 10-6

EQUIVALENCE OF ANALYTICAL CODES

(SM-2 Experimental Core - 68°F - Water Reflector - All Fixed Elements)

<u>Code</u>	<u>K_{eff}</u>	<u>ρ (%)</u>
PDQ-XY	1.155	13.4
VALPROD	1.153	13.3
FINK-1	1.153	13.3

TABLE 10-7

REACTIVITY OF VARIOUS CONFIGURATIONS OF THE SM-2
EXPERIMENTAL CORE - 68°F

<u>Core Configuration</u>	<u>Calculated Reactivity (%ρ)</u>	<u>Measured Reactivity (%ρ)</u>
Seven Control Rod Fuel Elements, SS Reflector	13.0	11.5 /
All Fixed Fuel Elements, SS Reflector	14.0	13.5 //
Seven Control Rod Fuel Elements, Water Reflector	12.3	10.9
All Fixed Fuel Elements, Water Reflector	13.3	12.9 ///

/ This value obtained using total K_{excess} in cents and $\rho = 1 - e^{-0.0078 \times K_{\text{excess}}}$ and integral rod worth curve presented in APAE No. 54, Figure 4.4.

// This value obtained using $\Delta\rho$ due to substitution of fixed fuel elements of +2.0% ρ .

/// This value obtained $\Delta\rho$ due to steel to water reflector of - 0.6% ρ .

TABLE 10-8

REACTIVITY OF VARIOUS CONFIGURATIONS OF THE SM-2
REFERENCE CORE - 510°F - NO XENON

<u>Core Configuration</u>	<u>Calculated Reactivity (% ρ)</u>
Seven Control Rod Fuel Elements, SS Reflector	6.7
Seven Control Rod Fuel Elements, Water Reflector	5.9
All Fixed Fuel Elements, SS Reflector	8.0
All Fixed Fuel Elements, Water Reflector	7.1

TABLE 10-9

BEST ESTIMATE OF SM-2 REFERENCE CORE REACTIVITY

<u>Core Condition</u>	<u>Reactivity (% ρ)</u>	
	<u>Calculated</u>	<u>Measured</u>
68°F, Clean, 0% Burnup	12.3*	11.5 /
510°F, Clean 9% Burnup, No Xenon	6.7*	4.8 //
510°F, Eq. Xenon, 0% Burnup	4.2**	2.3**
510°F, Max. Xenon, 0% Burnup	2.4**	0.5**

* Calculated by PDQ code, corrected for flow divider loss of 96.6 cents or 0.7% ρ .

** Calculated reactivity changes due to equilibrium and peak xenon are 2.5% ρ and 4.3% ρ respectively.

/ Taken from integral worth curve of 7 rod bank, corrected for flow divider loss of 0.7% ρ .

// The measured hot to cold reactivity change is 890 cents or 6.7% ρ .

4. Plans for Next Period

A new non-uniform burnup calculation of the reference SM-2 core has been started. It is planned to make this calculation using both the CANDLE-2 code for the IBM-704 and the NUB-1 code for the IBM-650; both codes are one-dimensional. Calculations involving the input to the CANDLE-2 code have been initiated.

Both codes will be used in order to compare the techniques generally used at Alco (the NUB-1 code) to a somewhat more versatile method. The variation of the axial power distribution with lifetime should be more accurately determined by the CANDLE-2 code. The CANDLE-2 code will permit a multistep (new flux distribution at each time step) and pointwise burnup calculation of the lifetime of the core. At present, the NUB-1, code generally employs a one-step, multiregion burnup with fluxes averaged over each region.

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B. CORE THERMAL DESIGN

1. Progress (Task 2.0) - I. Beretsky

A modification to the thermal stress equation for fuel elements has been developed and is being applied to SM-2 and the test elements in the VBWR, MTR, and STR. In addition the element removed from SM-1 will be analyzed.

An attempt was made to get the loss of flow code, ART, operational. Difficulty was experienced due to lack of certain documents.

Analysis of two-dimensional temperature distributions is proceeding. An AP Note on the flow re-evaluation is in preparation.

2. Thermal Stress Analysis of the SM-2 Fuel Elements - Task (2.10) - I. Beretsky

A modification to the stress equation appearing in APAE Memo 197 has been developed. This equation accounts for difference between material properties of the side plate and meat regions of the fuel element. These differences are dependent on both temperature and composition. This modified stress equation is being used to compare the ratio of the SM-2 stress with that occurring in the VBWR, MTR, STR and SM-1 tests.

3. Loss of Flow Kinetics (Task 2.9) - I. Beretsky

Documentation on the write-up of the ART code to treat loss of flow kinetics has been obtained. However, the write-up is incomplete. This documentation referred to is only preliminary. The final documentation has been published by WAPD as TM-156. This IBM-704 code which deals with thermal transients is quite similar in most respects to ATBAC. The major changes have been to obtain greater flexibility and to simplify input preparation. This code utilizes the more up-to-date estimation of two-phase pressure drop and burnout heat fluxes. It also enables the engineer to run the data at various system pressures and inlet temperatures without undue burden involved on input preparation. Secondly, the major bulk of temperature dependent properties are utilized on tape from a prepared table instead of linear approximations as needed previously.

4. Final Flow Reference Data (Task 2.8) - L. S. Hecht

This information, in support of the flow re-evaluation reported in APAE Memo 235, will be published in the form of an AP Note.

This analysis will be the effect of SM-2 required flow against various parameters such as:

- (a) Revised power distribution (Ref. Tables 10-12, 10-13 of APAE Memo 235)
- (b) Rod bank position
- (c) Effect of 40 mil side plates
- (d) Increased meat width and subsequent increase of heat transfer area.

Secondarily, the effect of core manufacturing deviations on SM-2 core flow will be shown.

5. Two-Dimensional Temperature Distribution in SM-Type Elements and Stainless Steel Vessel (Task 2.10) (Task 2.11) - L. S. Hecht

With the use of the IBM-704 at the NYU-AEC Computing Center in New York City, the heat conduction code HOC was run for the following general cases:

- (a) SM-1 control rod fuel plates
- (b) SM-2 control rod fuel plates
- (c) SM-2 stainless steel vessel shell and flange (Ref. Dwg. AEL-521).

An investigation is in progress which consists of:

- (a) Evaluation of the effectiveness of the dead edge portion of the fuel plate as a heat transfer fin.
- (b) Determining the effect of heat transfer coefficients on temperature distributions in control rod fuel plates and reactor vessel.
- (c) Determining the effect of heat generation on control rod plate temperature distribution.

- (d) Determining the effect of coolant temperature on control rod plate temperature distributions.

All data will be presented in the form of isotherms plotted on a two dimensional mapping of the solid in question. The various two-dimensional mappings will be (a) control rod fuel plates at the hot mid-plane, (b) control rod fuel plates from the leading edge to the mid-plane, and (c) vessel shell including flange region with and without the flange bolts.

A comparison with one-dimensional calculations will be made for the case of the vessel shell.

In subsequent work the two-dimensional temperature distribution in the inlet and outlet nozzles as well as in the vessel cover will be determined.

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C. REACTOR VESSEL DESIGN

1. Progress (Task 3.0) - T. F. Connolly

Some modification to the control rod and control rod fuel element has been made to permit centering of the element in the tube and to improve flow distribution.

Calculations on the stationary fuel element indicate that the flow forces are great enough to keep the fuel element against the top doors at all times.

An SM-2 stationary fuel element for use in the SM-1 reactor has been designed and detailed for Task 4b.

The top doors of the core support structure have been redesigned to improve flow and pressure drop conditions.

The pressure vessel calculations have been re-examined and the 347 and 304 vessel drawings sent back to Dunkirk for a recheck on the cost estimates.

As of this date only one bid has been received on the closure test vessel.

2. Fuel Element Design (Task 3.2) - T. F. Connolly

The problem of centering the fuel element in the control rod tube and adding a deflector to improve flow conditions were presented at the same time. Some redesign on the yoke end of the control rod was necessary and is shown on AEL-445 Rev. B. The orientation lugs were moved from the sides of the yoke towards the center and continued down along the axis of the yoke to serve as a centering device, and a deflector, and to still maintain the proper orientation of the fuel element.

Some consideration and study is now being given to the yoke of the control rod to reduce fabrication costs. The yoke shown on the drawing was "hogged" out of a block of steel. If this piece is fabricated and welded out of four smaller pieces it should result in a better, less expensive piece.

Redesign work was necessary on the control rod fuel element to add a comb and improve the handle. This is shown on AES-419 Rev. A.

With two long fuel plates at the bottom end of the fuel element, provision for addition of the comb became slightly complicated since it is undesirable to weld the comb to the fuel plates. Using the same philosophy as with the stationary fuel elements, the fuel plates were cut off to equal length, the comb set in place, and two ribs added to center to hold the comb in place. The two ribs also permit centering of the fuel element without inducing any shock loads to the fuel plates, during loading or transfer.

The problem of the stationary fuel elements rattling around in the core has been investigated and thus far no easy or sure way has been found to eliminate this. A leaf spring added to the end fixture is one method proposed to hold down the elements, but because of the small work area this is not too attractive.

Calculations on the flow forces vs. weight of the fuel element have been made. The weight of the fuel element in water is approximately 13-1/2 lb. and the flow forces are 72 lb., which indicate that the fuel element will be forced against the top plate at all times. At approximately 48% flow the weight will overcome the flow forces and the fuel elements will rattle. Since the reactor will never operate at much below 95% full flow except during shutdown, an evaluation will be made to determine whether it is necessary to prevent element movement.

A complete design (with detail drawings) of an SM-2 element for the SM-1 reactor has been made for Task 4b testing. The SM-2 fuel plates were used as is with the side plates modified and fitted to SM-1 end boxes.

3. Core Support Structure (Task 3. 3) - T. F. Connolly

The top doors of the core support shown on AEL-547 structure have been redesigned to improve the flow and pressure drop conditions. In the first pass, 5/8-in. dia. holes were used to get the correct pressure drop. In the second pass 1/2-in. dia. holes were used to approach the pressure drop of the control rods. This design has been assigned to Task 6 for incorporation into that test program.

Drawing AES-410 shows the relationship of the fuel and control rods in SM-1, SM-1A, PM-2A, and SM-2. This drawing indicates that the SM-2 fuel plates could be successfully used in any of these reactors.

4. Reactor Vessel (Task 3. 5) - T. F. Connolly

The drawings of the 347 and 304 stainless steel vessels have been returned to our Dunkirk Plant for rechecking the quotations.

5. Closure Test Rig (Task 3.9) - T. F. Connolly

Bids have been solicited for this item, with all replies not yet received.

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LAYOUT	
ALCO PRODUCTS INC.	
SUNSHINE, NEW YORK	
STEEL ERECTORY DEPT.	
MODEL	STEEL ERECTORY DEPT.
SCALE	1/4" = 1'-0"
DATE	12-15-54
AUTHORITY	ENGINEER
DATE	12-15-54
DATE	12-15-54
TITLE	
COVER, CORE SUPPORT	
LAYOUT NO. BEL-547	

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DATE	REV	REVISION	RECORD	AUTH.	DR.
1-4-60	A	SUPPRESSOR ADDED			
		TO FLUX, DIST. Wg			
		WAS 1/16, 1/8, 1/4			
		1/8 WAS 1/16

SM-1 & SM-1A

		MIN.	NOM.	MAX.
A	STATIONARY ACTIVE	21	21 3/4	22 1/2
B				
C	CONT. ROD ACTIVE	21 1/2	22	22 1/2
D				
E	ABSORBER ACTIVE	20 13/16	20 13/16	21 5/16
F	GAP (SCRAM) BOTTOM	-27/64	5/64	37/64
G	GAP (FULL OUT) BOTTOM	-27/64	5/64	37/64
H	GAP (SCRAM) TOP	-11/64	21/64	53/64
J	GAP (SCRAM) TOP	-11/64	21/64	53/64

		MIN.	NOM.	MAX.
A	STATIONARY ACTIVE	21	21 3/4	22 1/2
B				
C	CONT. ROD ACTIVE INCLUDING FLUX SUPPRESSOR	21 1/4	22	22 3/4
D	FLUX - (CONTROL ROD)	3/4	7/8	1
E	ABSORBER ACTIVE	20 13/16	20 13/16	21 5/16
F	GAP (SCRAM) BOTTOM	-27/64	5/64	37/64
G	GAP (FULL OUT) BOTTOM	-27/64	5/64	37/64
H	GAP (SCRAM) TOP	-11/64	21/64	53/64
J	GAP (SCRAM) TOP	-11/64	21/64	53/64

PM-2A

		MIN.	NOM.	MAX.
A	STATIONARY ACTIVE	21	21 3/4	22 1/2
B				
C	CONT. ROD ACTIVE INCLUDING FLUX SUPPRESSOR	21 1/4	22	22 3/4
D	FLUX - (CONTROL ROD)	3/4	7/8	1
E	ABSORBER ACTIVE	20 13/16	20 13/16	21 5/16
F	GAP (SCRAM) BOTTOM	-37/64	1/64	53/64
G	GAP (FULL OUT) BOTTOM	-27/64	1/64	43/64
H	GAP (SCRAM) TOP	-27/64	3/64	37/64
J	GAP (SCRAM) TOP	-27/64	3/64	37/64

SM-2

		MIN.	NOM.	MAX.
A	STATIONARY ACTIVE INCLUDING FLUX	22	22 1/2	23
B	FLUX SUPPRESSOR - (STAT.)	1/16	1/2	9/16
C	CONT. ROD ACTIVE INCLUDING FLUX	21 1/2	22	22 1/2
D	FLUX SUPPRESSOR - (CONTROL ROD)	1/16	1/2	9/16
E	ABSORBER ACTIVE	20 13/16	20 13/16	21 5/16
F	GAP (SCRAM) BOTTOM	0	1/4	1/2
G	GAP (FULL OUT) BOTTOM	0	1/4	1/2
H	GAP (SCRAM) TOP	-1/16	1/16	13/16
J	GAP (SCRAM) TOP	-1/16	1/16	13/16

LAYOUT	
ALCO PRODUCTS INC.	
SCHECTADY, NEW YORK	
ATOMIC ENERGY DEPT.	
MODEL	STARTED TYPING 12-20-59
REFERENCES	12-20-59 (230-23)
AUTHORITY	12-20-59 (230-23)
TITLE	
FUEL LOCATION IN SM-1, SM-1A, PM-2A & SM-2	
LAYOUT NO. AES-4101	

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D. METALLURGICAL DEVELOPMENT PROGRAM

1. Progress (Task 4. 0) - R. D. Robertson

Approval has been received for two of the expanded programs submitted previously under this task. Work has been initiated on designing and material procurement for testing SM-2 full size elements in SM-1 and a reduced size element in the WTR. This work is reported under Task 4a and 4b.

The report on the effects of neutron irradiation on the properties of structural materials is being published.

The testing program for stud and nut materials has been finalized. Actual testing will be initiated when materials are received.

Progress at BMI is covered in the next section of this report.

2. Preliminary Evaluation and Planning of Metallurgical Program (Task 4. 1) - A. S. Wilder

Verbal approval has been received for testing two fuel elements made to SM-2 concepts, in the SM-1. All further work on these elements will be reported as Task 4b of this program.

Loop testing of SM-2 fuel plates in a test reactor was investigated, beginning in December, 1959. A letter authorizing this testing has been received. All further work on these test assemblies will be reported in Task 4a.

A request for testing an SM-2 element in the MTR or ORR is included in the expanded program.

3. Type 347 Stainless Steel Investigation (Task 4. 2) - R. W. Kelleman

Replies have been received from inquiries made to several steel mills regarding the procurement of Type 347 and Type 304 SS with restricted chemistries. The mills interested in supplying these special grades of steel include Allegheny-Ludlum Steel Corporation, Universal Cyclops Steel Corporation, and Crucible Steel Company. The three mills indicated that they can guarantee Type 347 SS with 0.025 max. cobalt and 0.01 max. tantalum in the sizes specified, and Type 304 SS with 0.04 max. cobalt.

4. Discrete Burnable Poisons (Task 4.3) - R. W. Kelleman, C. R. Bergen

Work on this testing program has been suspended to await approval of the expanded SM-2 program.

5. Pressure Vessel Material Selection (Task 4.4) - R. W. Kelleman

The report covering the effects of neutron irradiation on the properties of structural materials is in final stages of preparation. It is expected to be issued in February.

METALLURGICAL DEVELOPMENT PROGRAM
BATTELLE MEMORIAL INSTITUTE

Development of Fuel-Element Systems

The reference fuel element contains a core of approximately 24.5 w/o UO_2 and 1.2 w/o ZrB_2 dispersed in Type 347 prealloyed stainless steel plus a short (1/2 in.) end section of suppressor material containing approximately 17 w/o Eu_2O_3 dispersed in an elemental 18-9 stainless matrix. This 0.030-in. -thick core is clad with 0.005-in. thick Type 347 stainless steel. Forty fuel plates containing natural or depleted UO_2 are being fabricated for welding studies. Ten of these plates have been completed and 15 plates are in various stages of completion. These plates are not being prepared by a standard procedure, but include minor variations which will provide useful information for establishing fabrication specifications. Small-scale specimens are also being used to evaluate fabrication techniques.

The fabrication of 28 fuel plates containing fully enriched UO_2 has been previously reported. Twenty of these plates were sent to Alco for critical assembly tests and eight of the plates were retained at Battelle for evaluation studies. Three of these plates were cut up for boron and uranium analyses with five 1-in. -sq sections being obtained for uranium analyses. The boron analyses and estimated boron losses are listed in Table 1. The uranium analyses and metallographic examination are listed in Table II. It should be noted that in all cases boron loss is considerably less than the 10 w/o excess boron added; thus, based on these values, boron content could be readily maintained within ± 5 w/o of the total boron added. It should also be pointed out that with no boron loss, large variations (± 10 w/o) are possible from specimen to specimen even though dimensional tolerances are maintained.

Other than close control of fabrication procedures, it may be also possible to control boron loss by the use of metal-coated ZrB_2 particles. Three batches of ZrB_2 with coatings of 2-3, 4-5, and 7-8 microns thick niobium have been prepared by the fluidized bed process, and fuel specimens containing these powders have been prepared for roll cladding. The coated powders are being tested by X-ray diffraction, metallography, chemical, and vacuum fusion analyses. The first batch showed that no reaction occurred between the niobium and the ZrB_2 , and little or no oxygen or other impurities were picked up during the process. Powder containing tungsten coatings is now being prepared.

Other studies in progress include the evaluation and specification of stainless steel powder. Specimens containing several special powders produced by Vanadium Alloys have been prepared but have not been evaluated.

TABLE 1
BORON ANALYSIS ON ENRICHED SM-2 FUEL PLATES

Plate	Section (a)	Calculated Boron Per Section, wt mg	Boron Analyses After Fabrication		Boron Change, w/o
			w/o	wt mg	
EAR-2	A	8.62	0.158	8.69	+0.81
	B	8.50	0.153	8.18	-3.77
	C	8.84	0.161	8.95	+1.25
	D	8.68	0.157	8.58	-1.15
	E	8.44	0.149	8.07	-4.39
	Average	8.62	-	8.49	-1.51
EAR-21	A	8.68	0.143	7.98	-8.06
	B	8.68	0.148	8.01	-7.74
	C	8.50	0.157	8.29	-2.48
	D	8.53	0.152	8.15	-4.45
	E	8.68	0.148	8.01	-7.74
	Average	8.61	-	8.09	-6.04
EAR-22	A	8.90	0.169	9.34	+4.95
	B	8.82	0.149	8.12	-7.95
	C	8.76	0.144	7.74	-11.40
	D	8.76	0.147	7.98	-8.90
	E	8.81	0.170	9.46	+7.39
	Average	8.81	-	8.54	-3.06

(a) 1 by 1-in. sections cut on a diagonal traverse across the core area.

TABLE II

<u>Plate</u>	<u>Section</u>	<u>Uranium Analysis, w/o</u>
EAR-2	F	15.5
	G	15.3
	H	15.3
	I	16.1
	J	16.3
EAR-21	F	15.5
	G	16.2
	H	16.1
	I	16.3
	J	15.2

The radiographs of the full-scale fuel plates revealed streaking, indicative of segregation of the UO_2 . This varied from plate to plate with some of the plates exhibiting a homogeneous structure. It is believed that the spherical UO_2 is more subject to particle segregation during the powder mixing operations. Studies are now being conducted to develop powder blending techniques that will prevent powder segregation. These include an evaluation of various binders and determination of the effects of variation in amounts used and method for introducing the binders. Preliminary results indicate that segregation of powders can be prevented by control of the powder blending process.

Development of Control and Suppressor Materials

A dispersion of Type 347 stainless steel and Eu_2O_3 is reference control and suppressor material for the SM-2. An evaluation study is being conducted to determine the optimum method of preparing the Eu_2O_3 particles sizes and fabrication techniques for the dispersion. The specimens prepared are being evaluated by corrosion testing and metallographic examination. A corrosion specimen, containing 33 w/o Eu_2O_3 prepared by the KAPL method of sizing particles and dispersed in a 71-18-11 matrix, was corrosion tested at Alco Products, Inc., and failed within a 24-hr test period. A duplicate set of specimens containing Eu_2O_3 sized by the same process was fabricated. These specimens were defected by a 0.040-in. hole through the cladding and corrosion tested in 600 F static degassed water at Battelle. This set of specimens also failed within a 24-hr test period verifying the previous results. The Eu_2O_3 incorporated in these specimens

analyzed 0.01 w/o silicon and the dispersion was encased with a low silicon barrier foil to prevent possible reaction with the wrought Type 347 stainless steel cladding. Corrosion specimens were also fabricated containing Eu_2O_3 contaminated with approximately 10 w/o silicon. These specimens failed at approximately the same rate as did the low silicon specimens. Additional specimens fabricated by the same techniques but containing commercially available minus 100 mesh, air-fired Eu_2O_3 have shown no weight gain or dimensional changes after a 24-hr corrosion test. Similar specimens have been prepared for corrosion testing at Alco. These tests will be more extensive than the preliminary screening tests run at Battelle. All specimens to date have contained an 18-11 chromium-nickel-iron matrix and were sintered in a hydrogen atmosphere furnace. Results of the corrosion tests are reported in Task 5.0.

Irradiation of Test Specimens

The three noninstrumented capsules, designated BMI-32-1, -2, and -3, were inserted in core positions of the MTR at Cycles 126, 127, and 128, respectively. As of the end of Cycle 136 (February 15), BMI-32-1 had accumulated an estimated maximum specimen fission burnup of 54 per cent; BMI-32-2 had accumulated an estimated average specimen fission burnup of 59 per cent; and BMI-32-3 had accumulated an estimated average specimen fission burnup of 42 per cent. These burnups are based on reactor quoted fluxes, nuclear mockup capsules and BMI dosimeter readings.

The first instrumented capsule, BMI-32-4, was inserted in an ETR beryllium reflector position for Cycle 20 which started on September 12. Although the auxiliary electric heaters have failed, specimen temperatures at full reactor power have remained close to 600 to 650 F (calculated, based on thermocouple temperatures of 400 F). One of the original four thermocouples is still operative. During December, the ETR did not operate until the latter part of the month. Cycle 24 accumulated 975 MWD and Cycle 25 which started about December 24 is scheduled to continue until January 10, 1960. At the end of Cycle 24, Capsule BMI-32-4 has accumulated 5842 MWD at an estimated average fission burnup of 17 per cent, based on quoted fluxes.

The second instrumented capsule, BMI-32-5, is assembled and ready for shipment. It is similar to BMI-32-4 in all respects except that a revised heater element-to-lead out wire connector is used. It is believed that this connector eliminates some of the possible causes of heater failure in BMI-32-4.

A theoretical analysis of heat transfer and temperature distribution in the BMI-32 capsule design has been completed. This analysis was conducted jointly with Alco utilizing the IBM-704 Data Processing Installation at New York University. The results are now being compiled.

IRRADIATION OF SM-2 TEST ELEMENT IN WTR

1. Progress (Task 4a) - E. W. Mangum

We have been directed to design, fabricate, and irradiate a test element assembly in the WTR. As a result of a design study, the fuel plate active core dimensions have been finalized as follows:

Thickness - $0.030'' \pm 0.001''$

Width - $1.490''$ min.

Length - $16.0'' \pm 1/2''$

A three-plate element will be used unless it can be shown that sufficient flow velocity can be obtained with a four-plate element.

2. Design - B. J. Byrne

The fuel plate dimensions and the element concept are the outgrowth of thermal, hydraulic, and nuclear analysis on different test element designs which would fit in the WTR test thimble as shown on Drawings AES-433 and AES-435. The plate width was chosen such that a four-plate element could be tested, if desired.

The boxed element concept was chosen in order to simulate SM-2 flow conditions on the outer plates.

The 16-inch active core length was selected since this would allow the use of a flux suppressor on each end of the fuel plates if required, and would permit testing of two fuel elements.

3. Future Work - B. J. Byrne, E. W. Mangum

Further study is being made of the number of flux suppressors per fuel plate, the number of fuel plates per element, the number of fuel elements to be tested and box configuration.

A study is now being made to determine the best method by which the test element can be held in position within the box.

4. Test Schedule - E. W. Manqum

Battelle Memorial Institute has been instructed to proceed with fabrication of the active core compacts based on the above dimensions and SM-2 reference fuel plate core composition.

Completion of fabrication of the test section is scheduled for March 11, 1960.

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IRRADIATION OF SM-2 ELEMENTS IN SM-1

1. Progress (Task 4b) - A. S. Wilder

We have been directed to fabricate two SM-2 elements for irradiation in SM-1. The fuel elements are scheduled to be inserted in SM-1 Core I in May 1960. The adaptation of the SM-2 fuel plate bundle to SM-1 is shown in Drawing R9-50-1112.

2. Fuel Plates - A. S. Wilder

Battelle Memorial Institute has been given instructions to proceed with fabrication of the fuel plates. One of the elements will be composed of 18 reference SM-2 fuel plates. The other element will have two plates each of nine different concepts, as given in Table 4-1.

At least one other type of commercial fuel material will be incorporated into the program, if delivery of this material can be made in time for fuel plate fabrication. In this event, the list of test concepts will be revised.

3. Type 347 Stainless Steel - A. Wilder

Reference SM-2 Type 347 stainless steel has a restriction in chemistry of 0.025 cobalt and 0.01 tantalum. Scheduling requirements have made it mandatory that a special heat of steel be fabricated by BMI. Procurement of the special materials to pour the heat of steel was initiated by BMI on January 25, 1960. BMI will roll fuel element side plate stock from this material, as well as cover plate and picture frame stock for fuel plate fabrication.

4. Fuel Element Welding - R. A. Shaw

Welding of fuel plates into elements will be done at Alco starting April 1, 1960. Prior to that date Alco will weld dummy assemblies to insure that the jiggling and welding procedures are satisfactory.

5. Side Plates and Combs - R. A. Shaw

Side plate and comb machining and grooving will be done by Raloid, Inc., which has done similar work for Alco in the present SM-2 program.

6. Fuel Element End Boxes, Springs, Pins and Attachment of End Boxes to Elements - R. A. Shaw

Sylcor has been contacted concerning assembly of the end boxes to elements, and furnishing of accessory hardware. Sylcor has indicated that they will be able to do this work, as well as handle transportation of the elements from Alco to Sylcor and return. A quotation from Sylcor is expected by February 15, 1960.

7. Reactivity Check

Element reactivity will be checked at the Alco Critical facility.

TABLE 4-1

FUEL PLATES FOR ONE SM-2 ELEMENT

<u>Plate Designation</u>	<u>Oxide</u>	<u>Clad</u>	<u>Assembly</u>	<u>Burnable Poison</u>	<u>Suppressor</u>
1. Core II Reference	Geneva	304 L or 347	Welded	B ₄ C	None
2. SM-2 Reference	Spherical	347	Welded	ZrB ₂	Eu ₂ O ₃
3. SM-2 Reference No Boron	Spherical	347	Welded	B-S.S. Picture Frame	None
4. SM-2 Reference	Geneva	347	Welded	ZrB ₂	Eu ₂ O ₃
5. SM-2 Green Com- pact	Spherical	347	Welded	ZrB ₂	Eu ₂ O ₃
6. SM-2 Reference	Spherical	347	Welded	Cb B ₂	Eu ₂ O ₃
7. SM-2 Reference	Spherical	347	Welded	Coated ZrB ₂	Eu ₂ O ₃
8. SM-2 Reference UN UN	UN	347	Welded	ZrB ₂	Eu ₂ O ₃
9. SM-2 Reference UN UN	UN	347	Welded	No Boron	Eu ₂ O ₃

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E. WELDED FUEL ELEMENT DEVELOPMENT

1. Progress (Task 5.0) - R. A. Shaw

A report covering the manufacturing procedure for stationary fuel elements has been written and is being reviewed prior to issuance. Subsequently, the report will be revised and re-issued to include the latest fixturing developed plus procedures for fabrication of control rods.

A report on the welding development program is being written.

An element with depleted annealed fuel plates was given a heat treatment at 800°F for 30 hours (derived by time-temperature parameter to represent life cycle at 600°F). Plate rippling and channel spacing measurements were taken before and after heat treatment. Rippling occurred on both outer fuel plates, caused by radiant heating from the furnace coils. No rippling can be seen on the 16 inner plates.

The static autoclave phase of the fuel element corrosion testing was completed. Results indicated that the TIG welded elements exhibited only slight general attack after 500 hour exposure in 635°F. simulated reactor coolant. Sections through crater cracks at ends of some weld zones in obsolete 0.030" thick plate elements showed possible extension of the cracks by corrosion. A study was begun to determine the extent of this cracking.

No crater cracks have been found in elements with 0.040" plates.

Testing of europium oxide dispersions in the form of clad and unclad specimens was begun.

2. Development of Welding Process (Task 5.1) - R. L. Harris

An element is being prepared for a thermal stress test. This element will contain water cooled side plates. Heating elements will be fixed between fuel plates to obtain a temperature gradient between side plates and fuel plates. Provisions are being made to take rippling measurements on both outside fuel plates during heating and at temperature. Thermocouples are being installed in 20 locations.

3. Corrosion Testing (Task 5. 2) - C. R. Bergen

Fuel Element Testing

Corrosion propagation of crater cracks and sub-surface micro-fissuring is of interest for two reasons:

1. General weakening of assemblies could result
2. Cracking may propagate into the fuel plate

The following test was performed to determine if corrosion could occur in weld cracks, micro fissures, hot short cracks etc. under simulated reactor coolant at 635°F.

An old welded dummy fuel element #2-2 (30 mil side plates) was obtained and given a dye penetration test which revealed about 85% crater cracking of the welds. One half of the fuel element was sectioned between each weld area giving six units. Units 1, 3 and 6 were reserved and the balance were placed in an autoclave in simulated coolant at 635°F. After cleaning and examining the crater cracks, only six of the welds under test did not have crater cracks. The specimens were exposed for 1,336 hours under degassed water. The final 493 hours of the time, the specimens were exposed to water of a nominal 8ppm oxygen content (nominal because water chemistry in a sealed autoclave is not readily followed). The purposes of the oxygen addition were to simulate such an accidental condition in a reactor and to accelerate possible stress corrosion, as oxygen is known to play a part in stress corrosion.

Following the exposure, four of the six welds which did not show cracks before exposure had developed cracks easily visible under 25 x magnification.

Of the unexposed specimens, 15 welds were sectioned through the cracked area and similarly 27 were sectioned after exposure. All specimens were carefully examined metallurgically. The following results were found:

1. Internal weld cracks not previously visible propagated to the surface in 4 to 6 welds.
2. One of the weld cracks in an untested specimen appeared to extend a short distance into the dummy fuel plate.
3. Based upon comparison with untested material which indicates broadening of existing cracks as a result of corrosion testing, propagation of weld cracks exposed to primary fluid can be expected.

These results confirm the necessity for sound uncracked welds in the as-fabricated condition and the use of 0.040" side plates as a requisite to obtaining sound welds.

Europium Oxide Dispersion Testing

Testing of five clad specimens was completed during the month. These were as follows:

OR#3: Specimen prepared at Oak Ridge and drilled through, prior tested for 72 hours at 575°F.

BMI#2a: 0.056 inch thickness drilled through and sectioned.

BMI#2b: 0.056 inch drilled only.

BMI#3a: 0.085 inch drilled and sectioned.

BMI#3b: 0.085 inch drilled only.

The BMI specimens were prepared from Heavy Minerals Europium oxide which had been high-fired under conditions which differed from ORNL requirements in gas blanketing and in uniformity of heating. The defects (i. e. drilling and sectioning) were made at Alco.

The ORNL specimen OR#3 was sectioned upon arrival and exposed to simulated coolant at 575°F. for 16 hours to verify the results found during prior testing. As expected, there was only a slight weight loss (7.2 mg) attributed to the leaching of europium from the surface exposed during the sectioning. The specimen was then returned to the autoclave for 8 hours at 635°F during which time a gain of 5.8 mg occurred. During a subsequent one month exposure, at 635°F., the specimen gained 0.6 mg weight. This weight change is quite small and could be considered insignificant except for the following evidence:

Metallographic examination of the specimen after testing showed several voids in the dispersion of the form of fine "cracks." It was indicated that these were the result of exposure to the water, by the fact that there were more cracks at the surface exposed to the water and less cracks at a distance. The cracks, or voids, widened at the exposed surface and were concentrated in the middle third of the dispersion.

The BMI specimens were exposed for 166 hours at 635°F. Weight gains of up to 0.3% were found with swelling amounting to as much as 10

percent of the thickness. The swelling appeared to follow the side frame boundary. Metallographic examination showed poor bonding of matrix to the barrier foil, especially at the side frames. In specimen 2A, cracking of the matrix attributed to swelling of the Eu_2O_3 was visible.

4. Future Work - R. A. Shaw and C. R. Bergen

Welding Development

A second element will be heated to 800°F . at a slower rate of heating and also with insulation to obtain uniform heating, to determine if rippling was caused by uneven heating.

Work is being initiated to develop a better procedure for welding control rod absorbers. Seven absorbers will be welded.

Corrosion Testing

Testing of Eu_2O_3 compacts will continue as these become available.

Dynamic corrosion testing of welded fuel elements will be delayed until after June 1, 1960 due to prior commitments of the loop.

A report will be written covering the corrosion testing for the fuel element welding program.

F. HYDRAULIC ANALYSIS AND TEST

1. Progress (Task 6.0) - W. M. S. Richards

A summary report of the thermal stress testing performed to date has been issued. Preparations for flow testing the latest stationary element design are proceeding.

Flow testing of the control rod configuration has been satisfactorily concluded. Design and component selection for the full scale testing is roughly 75% done. Instrumentation comprises the major part of the remaining work.

2. Thermal Stress Testing of Fuel Plates (Task 6.2) - J. A. Christenson

During the report period, APAE Memo No. 239, "Thermal Stress Testing of SM-2 Fuel Elements" was issued. This report covers all the work performed under this sub-task to January 1, 1960.

Work has been initiated to develop a new method for obtaining valid specimens for future testing. Previous tests have been invalidated by excessive specimen distortion prior to testing.

3. Single Element Flow Testing (Task 6.4a) - J. A. Christenson

a. Stationary Elements

Recent design changes to the basic SM-2 stationary fuel element have made the test mock-up unsatisfactory. Major emphasis during the report period has been placed on the design and procurement of end fixtures for the test fuel element which suitably duplicate the latest reference design. The end fixture being duplicated is shown as Type C, Drawing R9-13-1017 in Task 3, APAE Memo 235.

The next core support configuration to be tested will have a bottom plate design as shown on Drawing AEL-523 (in Task 3, APAE Memo 235), and a top plate design as shown on Drawing AEL-547 in Task 3, this report.

b. Control Rod Assembly

New instrumentation has been installed in the control rod fuel element and four tests have been performed during the report period. The new instrumentation features velocity measuring probes in each of the element flow

channels, from which a more comprehensive picture of the flow pattern is obtained than previously.

(1) Repeat Test of Previous Modification

The first test using the new instrumentation was made on the control rod as modified for previous tests (the semicircular cutouts on the front of the two long fuel plates were eliminated). Channel flow distribution during this test is shown in Figure 6-1. Maximum variation from average for this test was + 8% and -13.5% at 350 gpm flow rate. Thermal design calculations are based on a maximum control rod flow deviation of $\pm 12\%$. Since this is based on reactor conditions where as yet unknown plenum effects are anticipated, it is felt that these test results must show substantially better distribution than this $\pm 12\%$.

(2) Yoke Modification

Previous testing indicated insufficient flow in the outer channels. To improve flow, deflectors or turning vanes were installed in the yoke section of the control rod Drawing AEL-445, showing these deflectors, is presented in Task 3 of this report. Channel flow distribution during this test is shown in Figure 6-2. The improvement in outer channel flow over the previous test is obvious. Maximum deviation from average was + 7.8%, - 8.1% exclusive of channels O and P which were felt to be questionable.

(3) New Fuel Element

The fuel element used during the above testing contained removable fuel plates for instrumentation purposes. It was suspected that the loose fuel plate between channels O and P distorted during the test to cause the abnormal distribution shown in the previous test (Figure 6-2). To confirm this suspicion, the test element was replaced with a fully welded element recently fabricated for Task 11 testing. Results of this test are shown in Figure 6-3, again at 350 gpm total rod flow. Distribution was within + 7.0%, - 7.4%.

(4) Pressure Drop Testing

All pressure drop data previously obtained on the SM-2 control rod was taken with all of the velocity measuring instruments and leads installed within the rod. It is felt that this additional material could be causing a significant increase in the static pressure drops being measured. To resolve this question, the

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FLOW DISTRIBUTION 5M-2 CONTROL ROD FUEL ELEMENT

TOTAL ELEMENT FLOW - 350 G.P.M.

FIGURE 6-1

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FIGURE 6-2

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FLOW DISTRIBUTION
 SMA-2 CONTROL ROD FUEL ELEMENT
 WITH ENTRANCE DEFLECTOR, FULLY WELDED ELEMENT
 TOTAL ELEMENT FLOW - 350 GPM

FIGURE 6-3

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majority of velocity measuring instruments have been removed and testing is in progress. Preliminary results indicate a measurable reduction in static pressure drop through the rod, but complete results are not yet available for analysis.

No further control rod testing in this rig is anticipated. It is felt that the flow distribution shown in Figure 6-3 is fully adequate to meet SM-2 design requirements. This is substantiated by two observations:

- (a) Referring to Figure 6-3, it can be seen that channels on the right hand side of the figure have higher flow rates than corresponding channels on the left side of the figure. This phenomenon has been noticed in previous tests performed in the same test section and is thought to be caused by a maldistribution of flow entering the test section. Therefore, since the control rod is symmetrical about the center channel, if flow through channels B and P are averaged and D and N are averaged, maximum deviation from element average becomes $\pm 5.0\%$.
- (b) Outer channels (A and Q), where flux peaks occur, receive essentially average element flow rates as shown in Figure 6-3. This is a considerable improvement over the condition shown in Figure 6-1 where channel A was 13.5% below average.

4. Full Scale Rig (Task 6.4) - D. England

A major fraction of the design effort for the entire loop, and the vessel in particular, has been completed. Checking and revision of detailed drawings are now in progress. Review of these drawings by Tasks 3, 4 and 6 personnel is expected by February 20, 1960.

The vessel and loop will be of carbon steel construction. Rust protection will be accomplished by commercial pipe coating for the loop and a cadmium coating on the vessel and core. The vessel and core is full scale size with a reduced pressure of 125 psi and temperature of 200°F.

Instrumentation is planned to determine element to element flow distribution as well as flow divider deflection and shield passage distribution.

An investigation to determine whether some surplus government pumps and motors can be used has been conducted with negative results. To use

this equipment, existing equipment would have to be modified to loop requirements at a cost exceeding the original cost of equipment properly matching the specifications. Final quotations for ordering the pump and motor, pipe, instruments, and supporting machinery have been solicited.

Our engineering laboratory has been forwarded layout drawings for both a floor and pit-mounted loop configuration together with water, drainage and power requirements. Site selection and test needs are being coordinated by the engineering laboratory.

Fabrication of the dummy core by our welding laboratory is scheduled to start on February 8, 1960. Fabrication of jig and fixturing equipment and machining of the fuel plates are completed.

G. CRITICAL EXPERIMENT PROGRAM

1. Progress (Task 7.0) - W. J. McCool

During this report period no reactor experiments were performed. The report on all SM-2 critical experiments performed to date was issued as APAE 54, "SM-2 Critical Experiments, CE-1." During this period design work was completed on a modified core support structure and flow divider. It is expected that measurements of flow divider effects and flux suppressor effects will be performed during February. In addition, the reactivity effect of substituting reference SM-2 fuel plates fabricated by BMI will be determined.

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H. CONTROL ROD DRIVE DEVELOPMENT

1. Progress (Task 11.0) - G. M. VandeMark

The prototype control rod drive mechanism has been completed. This design is based on a 23.2 ft. static pressure drop across the control rod. If it is found that the pressure drop exceeds 23.2 ft., modifications will be made to the test loop and drive mechanism during prototype testing.

All components of the prototype drive have been received in the laboratory. They have been inspected to the detail drawings and any discrepancies found have been corrected. At the present time the drive mechanism is being assembled and photographs are being taken of important components susceptible to wear.

All material required for the fabrication of the test loop has been received and approximately 90% of the erection has been completed. A minor problem was encountered in alignment of the primary loop piping, which has been resolved. All piping should be connected and the loop ready for hydrostatic testing during the week of February 8, 1960.

Since the possibility of a high static pressure drop across the SM-2 control rod exists, and investigation has been made to determine the extent of modification required to the test loop and drive mechanism to incorporate the hydraulic balanced control rod. The modifications required are shown on Drawing AEL-560. Drawing R9-50-1065 shows the present autoclave design, and is included for reference.

The modifications required consist basically of plugging the present flow holes around the pinion support, modifying the present pinion support and drive mounting flange to incorporate the larger rack, and adding a transfer tube to the autoclave.

2. Improved Accessibility of Drive Components (Task 11.1) - G. M. VandeMark

No further work has been done on this phase of Task 11.

3. Seal Leakage (Task 11.2) - G. M. VandeMark

The seal assembly has been completed, and seal leakage tests will be run with prototype control rod drive testing.

4. Acceleration Device Design (Task 11. 3) - G. M. VandeMark

No further work has been done on the acceleration device with the exception of the hydraulic balance as discussed above.

5. Clutch Development (Task 11. 5) - G. M. VandeMark

No further work has been done in connection with this portion of Task 11 since the previous review period.

An APAE Memo documenting the clutch tests has been prepared and is nearly ready for publication.

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