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**SM-2
REACTOR CORE AND VESSEL
REVIEW REPORT**

August 25, 1959 to December 14, 1959

J. G. Gallagher - Project Engineer

Issued: December 24, 1959

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INTRODUCTION

This review report contains a summary of the progress made since the last Review Meeting (September 16, 1959). A formal review of the SM-2 reactor core and vessel program will be held at Schenectady, N. Y., on January 5, 1960.

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PROGRESS SUMMARY

1. The most adverse power distribution has been revised based on a comparison of PDQ calculations and measurements made during the SM-2 flexible experiments. A review of the basic nuclear data and calculational models employed in the SM-2 nuclear analysis was made. A comparison between initial reactivity, hot-to-cold reactivity change and xenon reactivity with experiment was made. (Task 10).
2. Based on the revised power distribution developed in Task I, the core flow requirement has been reestimated to be 7800 gpm. As part of this reevaluation a review of each factor affecting volume flow requirement was completed. The revised volume flow through the control rod assemblies is 331 gpm and the maximum volume flow through a fixed element is 316 gpm. This volume flow is based on uniform flow through all fixed elements in the second pass. (Task 2).
3. A Type 304 stainless steel vessel drawing has been prepared and sent out for cost estimates. Tentative designs of the core support and fuel element structure have been prepared and evaluated for pressure drop and flow distribution. (Task 3).
4. The irradiation specimens in the MTR core were removed and reinserted because of difficulty with another experiment. The burn-up in the MTR capsules ranges from 18 to 50% U-235. The heaters failed on the instrumented capsule in the ETR. Additional ETR capsules have been delayed pending development of an explanation of the failure or suitable assurance failure will not recur. Type 403 stainless steel has been selected as reference stud and nut material. Type 304 stainless steel has been selected as reference pressure vessel material. (Task 4).
5. The TIG process has been selected as the reference method for welding elements. No crater cracking has occurred since changing from 0.030" to 0.040" side plates. Examination of an element welded with two depleted fuel plates with canted cores indicated that a minimum of 0.050" fuel plate dead edge is adequate. This results in an increase in heat transfer areas of approximately 7%. Preliminary specimens of Eu_2O_3 dispersions prepared by BMI were autoclave tested and showed severe swelling and distortion. Autoclave tests with an ORNL sample at Alco and ORNL indicated no swelling or distortion under the same conditions. Eu_2O_3 crystal structure appears to have an important bearing on corrosion resistance of Eu_2O_3 stainless steel dispersions. (Task 5).

PROGRESS SUMMARY (Cont'd)

6. Static deflection measurements indicate that a fuel element with cold rolled plates will have a deflection approximately 18% lower than annealed plates. Thermal deflection tests on a fuel plate cut from a TIG assembled element were inconclusive because the plate distorted when cut from the assembly. Vibration tests indicate no significant plate vibrations up to 164% of rated flow. Measurement of plate collapse on two elements indicated possible collapse in the range between 140% and 164% of rated flow. A leading edge comb indicated no collapse for the same flow range. Flow distribution and pressure drop tests were made for several core support structure configurations. Results indicate that a core support structure consisting of multiple small holes will probably give good distribution and the required pressure drop. Detailed design of the full scale water flow rig is progressing satisfactorily. (Task 6).
7. Mockup experiments on the SM-2 initial cold, clean and SM-2 mid-life cores were completed. Limited power distribution and flux distributions were performed in the clean mockup. The hot-to-cold reactivity change was measured by aluminum displacement as 890¢. The average B-10 and U-235 worth in the clean mockup were measured as 43 and 0.157¢ per gram, respectively. The reactivity effect of replacing control rod fuel assemblies by stationary fuel elements was measured in the clean mockup. Stuck rod positions were measured in the midlife mockup. (Task 7).
8. Approximately 25% of components for the prototype control rod drive mechanism have been received. All material required for fabrication of the test loop has been received with the exception of the test autoclave. Delivery of the autoclave is promised for the week of December 28. Test loop erection is approximately 30% complete. (Task 11).

DESIGN PARAMETERS

SM-2 Core and Vessel

Geometrical - Core

Assembly Drawing	AES-319
Core Configuration	7 x 7 - corners missing
Cell Size, in.	2.9375 x 2.9375
Active Core Height, in.	22
No. of cells	45
No. of control rod cells	7
Gap for flow divider, in.	0.218
Flow divider thickness, in.	0.094
Equivalent diameter with flow divider, in.	22.74

Geometrical - Elements

Assembly drawing - fixed element	R9-13-1017
Assembly drawing - control rod	AEL-445
No. of fuel plates	
Fixed element	18
Control rod element	16
Fuel plate meat thickness, in.	0.030
Cladding thickness, in.	0.005
Total fuel plate thickness, in.	0.040
Side plate thickness, in.	0.040
Plate width, in.	
Fixed element, in.	2.839
Control rod element, in.	2.589
Meat width, in.	
Fixed element (min.)	2.650
Active length, in.	
Fixed element	22
Control rod element	21.5
Total plate length, in.	
Fixed element	23.5
Control rod element	23.0
Length of europium flux suppressor, in.	0.50

Material Composition - Core

Fuel	Highly enriched UO_2
Burnable UO_2	Natural boron in ZrB_2
Wt. % UO in matrix	24.5
Wt. % natural boron in matrix	0.191
Cladding	Type 347 SS (0.025 w/o Co and 0.01 w/o Ta)
Absorber material	Eu_2O_3
Flux suppressor material	Eu_2O_3

DESIGN PARAMETERS (Cont'd)

Loading - Core

U-235 per plate, gm	
Fixed element	46.30
Control rod	40.56
B-10 per plate, gm	
Fixed element	0.08106
Control rod element	0.07102
Atom % U-235 in meat	5.2
Atom % B-10 in meat	0.21
 Total U-235, kg	 36.2
Total B-10, gm	63.4

Reflector

Drawing number	AES-341
Material	Type 304 S.S. (0.04 w/o Co)
Thickness, in.	8
Integrated Neutron Flux (NVT >1Mev), (neutrons cm ⁻²)	1.0x10 ²¹
Thickness (2200 psia design), in.	4-1/2
Volume, ft. ³	101
Weight, lbs.	31,300
Bolt and nut material	Type 403 S.S.

Nuclear Characteristics

Reactivities, %	
68°F, clean, 0% Burnup	11.5
510°F, clean, 0% Burnup	4.8
510°F, Eq. Xe., 0% Burnup	2.3
510°F, Max. Xe., % Burnup	0.5
68°F, No Xe., 22% Burnup	15.8

Bank positions, inches from bottom of core

68°F, clean, 0% Burnup	7.0
510°F, clean, 0% Burnup	11.3
510°F, Eq. Xe., 0% Burnup	14.1
510°F, Max. Xe., 0% Burnup	17.3
68°F, No Xe., 22% Burnup	4.6
Temperature coefficient (at 510°F), °F ⁻¹	-4.1 x 10 ⁻⁴
Core lifetime, MWYR	25.7
Control rod worth, (%)	
7 rod bank	22.8
6 rod ring	17.9

DESIGN PARAMETERS (Cont'd)

Maximum to average radial power (including local peaking)	2.10
Maximum to average axial power	1.84
Initial average thermal flux in core	2.0×10^{13}

Thermal - Hydraulic Characteristics

No. of coolant passes	2
Reactor power (reference), MW	28
Heat flux, BTU/hr-ft ²	
Average	1.68×10^5
Maximum	6.50×10^5
Burnout Ratio (Pass II)	2.63
Maximum surface temperature, °F	635
Maximum plate centerline temperature, °F	697
Inlet coolant temperature, °F	500
Outlet coolant temperature, °F	530
Total volume flow, GPM	7800
lbs/hr $\times 10^{-6}$	3.05
Maximum coolant flow through fixed elements, GPM	
First pass	218
Second pass	316
Coolant volume flow through control rods, GPM	331

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PLANS FOR NEXT PERIOD*

1. A new non-uniform burnup calculation of the SM-2 will be completed. Analysis in the interpretation of the SM-2 critical experiments will be made in order to further check the calculational model. A report covering the SM-2 nuclear analysis performed to date will be issued. (Task 10).
2. Stresses and temperature distribution throughout various reactor components will be established. The loss of flow behavior will be re-investigated for the final design utilizing a 704 code. (Task 2).
3. Core support structure, control rod and fuel element drawings will be finalized. Full scale closure test materials will be ordered. Final stress analysis for all core and vessel components will be largely completed. (Task 3).
4. All instrumented capsules for the ETR will be fabricated and inserted. Further full-size fuel plate rolling development will be carried out. Corrosion testing of defected Eu₂O₃ dispersion will continue. (Task 4).
5. An initial draft of manufacturing procedures for TIG-welding of fuel elements will be prepared. Dynamic corrosion testing of elements will be instigated. (Task 5).
6. All single element testing will be completed and engineering reports issued. All design work on the full scale water rig will be completed and materials ordered. Final hydraulic analysis of the core and vessel will be instigated. (Task 6).
7. The prototype control rod drive mechanisms will be completed and initial testing begun. The test loop will be completed. (Task 11).

* The next period ends March 1, 1960

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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.

A. CORE NUCLEAR DESIGN

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

By use of PDQ, an IBM-704 code, flux and power distributions and core reactivity were calculated and compared to measurements made during the SM-2 Flexible Critical Experiments. From this comparison, revised power distributions were calculated and given to Task 2 for the calculation of new flow rates. It was found that, in general, the revised peaking factors are lower than the previously published values.

A review of the calculational methods used in the nuclear analysis of the SM-2 core was made. The elastic and inelastic scattering files for iron, used in the MUFT III, IBM-650 code to calculate the fast nuclear parameters, were reviewed based on the latest available data. Use of the revised files in the calculation of the neutron age from a Po-Be source in various iron-water mixtures yields very good results as compared to experimentally measured values. Use of the revised files reduces the calculated reactivity of both the SM-1 and the SM-2 cores.

The use of various calculational models, including P_3 theory, P_1 theory, and Wigner-Wilkins averaged cross sections, upon the thermal parameters and thermal innercell flux distributions, were studied. It was found that the core reactivity is not a strong function of the model employed. The presently employed model, based on P_3 theory, offers a good compromise between the accurate prediction of core reactivity and innercell thermal flux distributions.

The latest estimate of the expected lifetime of the SM-2 core is 25.7 MWYR. The effect of coating the ZrB_2 particles with various materials was calculated; a tungsten coating of thickness equal to 30% of the particle volume yielded the greatest reactivity loss of 0.14%.

At the start of core life, the zero to equilibrium and zero to peak xenon reactivity changes were calculated to be 2.45% ρ and 4.23% ρ respectively.

It was found that reducing the fuel and boron loadings in the SM-2 control fuel elements not only reduced the core reactivity and power within the element, but resulted in increased power peaking in the vicinity of the element. The gamma heating throughout the SM-2 pressure vessel wall was calculated based on previous calculations made in a radial direction only.

2. Core Analysis (Task 10.1) - P. E. Bobe, E. F. Clancy, D. H. Lee

a. Xenon Reactivity Transients

Using the model developed in APAE Memo 38, a study was made of transient xenon parameters in the SM-2 by use of analog computer methods.* Assuming α , the xenon distribution factor, to have the same value for the SM-2 as for the SM-1, the reactivity changes due to xenon were simulated on the computer. At startup, the analog simulation shows that ρ_{xe} equals 2.45% and 4.23% for zero to equilibrium xenon and zero to peak xenon, respectively. These values were found to compare favorably with values previously reported and calculated by numerical methods.

b. Core Lifetime

The original non-uniform burnup calculation of the SM-2 was based upon a one-pass core at 440° F, 30 mil side plates, and with a thick steel reflector. This lifetime was calculated to be 28.1 MWYR. Since the original calculation, there have been several design changes, all of which affect the lifetime. A new non-uniform calculation has been delayed due to the high cost of performing the calculation and the possibility of revising the present calculational techniques to obtain a more accurate model. In addition, if additional funds are made available, it is possible that more sophisticated methods will be employed, for example, the TURBO, IBM-704 code.

However, a good estimate of the core lifetime may be made with knowledge of the effect of various design changes upon core reactivity. The effect of various design changes, based on results of the SM-2 Flexible Critical Experiments, are given in Table 10-1.

TABLE 10-1
EFFECT OF DESIGN CHANGES

Design	$\Delta \rho$ (%)	Δ (MWYR)
Flow Divider	-0.75	-0.5
440° F to 510° F	-2.1	-1.4
All water reflector	-0.60	-0.4
Thin steel skirt	-1.20	-0.8
30 to 40 mil side plates	-0.7	-0.5

* Lee, Dana H., "Xenon Reactivity Transients," AP Note 202, December 3, 1959

Therefore, the reference core at 510°F with 40 mil side plates, and with a flow divider has an expected lifetime of about 25.7 MWYR. A new non-uniform burnup calculation is being planned for a better estimate of the expected core lifetime.

c. Fuel Burnout Distribution in Element 45

The local burnout distribution across the width of fuel plate number 9 of fixed element 45 (Figure 10-1) was calculated corresponding to 27 MWYR operation of the SM-2. The method used was essentially the one presented in APAE Memo 199.* The formula used is as follows:

$$B_j^n(Z) = A' B(Z) F(X) F(Y) P(j)$$

$$B_j^n(Z) = \text{local burnup in a fuel element at position } (n, j, z)$$

$$n = \text{number of the fuel plate in the particular element under consideration.}$$

$$j = \text{section of the fuel plate. (The plate width was divided into quarters, A through D).}$$

$$A' = \text{value of the power on the centerline of the core as calculated by the VALPROD code.}$$

$$F(X), F(Y) = \text{Geometrical power factors which result from an attempt to separate a radial power distribution into X and Y components. **}$$

$$P(j) = \text{local side plate peaking factor obtained from } P_3 \text{ calculations.}$$

$$B(Z) = \text{average axial fuel burnout distribution.}$$

Since the most adverse burnup distribution was desired, the axial point corresponding to 15 inches from the top of the core in element 45 was chosen. The resulting factors are shown in Table 10-2. It should be noted that these calculations assume that the innercell flux distribution does not change with time. The maximum fuel burnup (Atom % U-235) is 92% for a 1/2 sq. in. area. This is an upper limit on burnup.

* McElligott, P. E., "Burnout Distribution in SM-1 Control Rod Elements, Fixed Element 57, and Absorber Sections at 10.5 MWYRS," APAE Memo 199, June 5, 1959.

** Byrne, B. J., "Power Distribution for APPR-Type Cores," APAE Memo 195, July 28, 1959

TABLE 10-2
LOCAL BURNUP IN PLATE 9 OF FUEL ELEMENT 45

j	A ¹	F (X)	F (Y)	P (j)	B (Z)	B _j ⁹ (Z)
A	1.675	0.970	1.00	1.08	0.526	0.923
B	1.675	0.954	1.00	0.97	0.526	0.815
C	1.675	0.913	1.00	0.97	0.526	0.781
D	1.675	0.898	1.00	1.06	0.526	0.841

d. Effect of Coating the ZrB₂ Particles

If the fuel plates are sintered in a hydrogen atmosphere instead of in vacuum, it may be necessary to coat the ZrB₂ particles with non-reactive material such as tungsten, chromium, niobium, etc. Coating thicknesses that would equal 10%, 20%, or 30% of the particle volume are being contemplated. As support to the metallurgical program preliminary calculations were made to determine the loss of core reactivity as a function of coating thickness and material. The total boron loading was held constant. These results are shown in Table 10-3.

TABLE 10-3
REACTIVITY LOSS (%) DUE TO COATING OF ZrB₂ PARTICLES

Coating Material	Coating Thickness *		
	0.10	0.20	0.30
Tungsten	0.046%	0.092%	0.138%
Chromium	0.009	0.018	0.027
Niobium	0.002	0.005	0.007

* Fraction of total particle volume.

e. Water Hole Flux and Power Peaking in the SM-1, Core II

To investigate the possibility of placing an element in the SM-1, Core II containing one or two SM-2 reference fuel plates, the thermal flux and power peaking in the vicinity of such an element was calculated. Due to limitations inherent in the VALPROD IBM-650 code, it was not possible to obtain the flux and power distribution in the vicinity of the SM-2 fuel plates. Therefore, cylindrical VALPROD's were run for the core at 440°F

both for the normal control fuel element and for a water hole in the center of the core.

Figure 10-2 shows the thermal flux to which the SM-2 fuel plates would be subjected. An SM-2 fuel plate in the center would be subjected to a flux peaking factor of about 6 to 1 over a normal SM-1 plate in a regular SM-1 element.

Figure 10-3 shows the large increase in power peaking next to the water hole. The peak to average (average in the core) power for the water hole is 3.1 as compared to the value of 1.7 next to a regular control fuel element.

f. Gamma Heating on the SM-2 Pressure Vessel

The gamma heat generation rate throughout the inner surface of the pressure vessel wall of the reference SM-2 reactor was calculated at 510°F near the end of core life. Since there were no IBM-650 codes available to determine the gamma heat generation in non-axial or non-radial directions, it was necessary to extrapolate the results found in the radial direction as presented in AP Note 188.* The heat generation rate along the inside pressure vessel wall was obtained by considering the effects of the increased amounts of reactor materials that attenuate the gamma flux from the core and secondary sources as a function of distance from the pressure vessel belt line. The calculations were made based on the important assumption that the ratio of the structural materials and water between the core and the vessel remains constant as a function of distance from the pressure vessel mid-point. This is very nearly the case from 6 to 8 inches from the top and bottom of the core.

At larger distances, the structural material is cut off and the water fraction between the core and the vessel increases rapidly. Since the attenuation coefficient of water is smaller than that of the steel, the gamma heat generation on the inner wall of the pressure vessel will be somewhat higher than those based on the assumption stated above. This is compensated to some extent by a decrease in the secondary (n, γ) source in the structural materials. A lack of knowledge of the thermal and fast fluxes at oblique angles to the core prevented an estimate of their effect on the heat generation on the pressure vessel wall. The results of these calculations are shown in Figure 10-4.

The gamma heat generation on the flange, assuming that the gamma attenuation is by water alone, was found to be 2.1×10^{-5} BTU/in³ sec. Since this is a conservative value, no correction for any effects by structural

* Clancy, E. F., "Gamma Heating in the Reference SM-2 Core and Reflector Regions," AP Note 188, October 15, 1959.

material is included. The gamma heat generation at the inner edges of the nozzles can be read from the graph.

g. Effect of Reduced Fuel Loadings in SM-2 Control Fuel Elements

The most serious power peaking occurs in the vicinity of the central control rod fuel element; therefore, this element fixes the flow requirements for the entire core. High flow through the control elements may be serious; therefore, as one possibility of reducing this peaking, the effect of reducing the fuel and burnable poison loadings of the control fuel elements was calculated. The radial power distribution and core reactivity was calculated by use of the VALPROD, IBM-650 code. A 10% and 20% reduction of U-235 and B-10 from the reference loadings were considered. The effect upon core reactivity is shown in Table 10-4. The effect upon the radial power distribution is shown in Figure 10-5.

TABLE 10-4
REACTIVITY CHANGE DUE TO REDUCTION IN U-235 AND B-10 LOADINGS

	$\Delta\rho$ (Central Rod Only)	$\Delta\rho$ (7 Control Elements)
Reference Loading	-	-
10% Reduction	- 0.058%	- 0.270%
20% Reduction	- 0.121%	- 0.559%

It is seen that a reduction in loadings causes a decrease in the power within the control fuel element but causes an increase in the peaking in the adjacent fixed fuel element; however, high flow through fixed elements is not too serious. As seen in Section 3 of this report (Task 10.2), the most serious peaking occurs in the adjacent fixed element, even in the reference core; therefore, reducing the loadings in the control fuel elements results in even more severe peaking and flow requirements in the fixed elements adjacent to the control elements. On the other hand, in the control fuel elements, 10% and 20% loading reductions result in 3.7% and 7.4% reductions in power peakings, respectively.

h. Review of Calculational Models

1. Cross Sectional Data for MUFT - III Input

The fast group parameters used in the modified two group theory are calculated by use of the MUFT-III, IBM-650 code with the P_1 Selengut-Goertzel slowing down approximation. Inelastic scattering from all neutron groups by stainless steel (or iron) and U-235 have been approximated by use

of the Weisskopf statistical model. Since this model implies large level densities in the excited nuclei, it is not applicable at incident energies below a few Mev's. In view of this, the inelastic matrix for Fe has been modified to account for scattering by individual levels in the low lying regions. In addition, the matrix has been changed in the higher energy regions in accord with more recent measurements of the nuclear temperature. The reactivity of various SM-1 cores were recalculated utilizing the new inelastic scattering matrix; the effect was to increase the calculated reactivity by about 0.2% ρ .

The new inelastic scattering matrix was also used to calculate the neutron age of Po-Be neutrons in various iron-water mixtures and compared to measured values.* However, the calculated ages showed poor agreement for high metal-water ratios (Table 10-5); therefore, the effects of re-evaluating the elastic parameters on the calculated neutron ages were investigated. The latest data of elastic scattering by iron was obtained from work done at KAPL.** The results of the calculation of the neutron ages in iron-water mixtures using a Po-Be source to the indium cut-off are shown in Table 10-5. It is seen that the agreement between the calculated and measured neutron ages is very good.

TABLE 10-5
NEUTRON AGES OF PO-BE NEUTRONS IN FE-H₂O MIXTURES

Metal/Water Volume Ratio	Measured Age (cm ²)	Calculated Age (cm ²) (Inelastic Change Only)	Calculated Age (cm ²) (Elastic and In- elastic changes)
0	55.4	54.7	54.7
0.5	56.2	40.8	54.5
1.0	58.8	42.9	58.4
2.0	72.1	49.9	68.0

The effect of incorporating both the elastic and inelastic modifications to iron upon the fast parameters and calculated reactivity of various SM-1 and SM-2 cores was also investigated. Table 10-6 shows a comparison of the fast group parameters, with and without the iron file modifications, for various SM-1 and SM-2 cores at 68 F.

It is seen from Table 10-6 that use of the modified Fe files for the

* Reier, M. F., F. Obenshain, R. L. Hellens, "The Ages of Po-Be Neutrons in Various Metal-Water Mixtures," Nuclear Science and Engineering, Vol. 4, 1-11 (1958).

** Private communication with E. F. Clancy, Alco Products, Inc., October, 1959.

MUFT-III code reduces the calculated reactivity. For the core studies, the reduction in reactivity is about 1.6% ρ and 1.1% ρ for the SM-1 and SM-2 cores respectively; part of this difference may be due to differences in the metal to water ratio of the two cores. This work will be issued as an AP Note.

TABLE 10-6
FAST GROUP PARAMETERS USING NEW Fe MUFT-III FILES

(Fixed Elements at 68°F)

Parameter	SM-1 Core *		SM-2 Core **	
	Old Value	New Value	Old Value	New Value
γ (cm ²)	32.914	35.074	33.233	35.90
D_F (cm)	1.3216	1.4167	1.3058	1.417
$\sum a_F$ (cm ⁻¹)	0.01018	0.01021	0.01387	0.01392
$\sum_{FF}$ (cm ⁻¹)	0.01308	0.01316	0.02055	0.02064
P	0.7466	0.7471	0.6470	0.6476
K_F	1.285	1.287	1.481	1.482
ρ (%)	15.4	13.9	26.32	25.23

* Reference SM-1 core (APAE 27) with "as specified" boron content.

** SM-2 Flexible Experiment Core with no boron. (18 plates per element).

2. Effect of Calculational Model and Neutron Temperature Upon Thermal Calculations

As part of the review of the calculational methods employed in the nuclear analysis of the SM-2 core, the thermal parameters and thermal flux distribution at 68°F within a fuel element were calculated by different analytical models and compared. The method presently used employs P₃ theory with cross sections averaged over a Maxwell-Boltzmann distribution evaluated at a hardened neutron temperature. For comparison, the flux and thermal parameters were calculated by:

- P₁ (diffusion) theory using hardened, Maxwell-Boltzmann averaged cross sections
- P₃ theory using Wigner-Wilkins averaged cross sections
- P₃ theory using non-hardened, Maxwell-Boltzmann averaged cross sections

The flux (parallel to the fuel plates) comparisons are shown in Figures 10-6, 10-7, and 10-8. The effect upon the thermal parameters and core reactivity for 45 fixed fuel elements is shown in Table 10-7.

It has been found from analysis of the SM-2 Flexible Critical Experiment that the measured innercell flux and power peaking exceeds that calculated by the presently used model. Therefore, the use of P_1 theory or the more sophisticated Wigner-Wilkins theory offers no advantage over the present model since both methods predict less peaking (Figures 10-6 and 10-7) and no appreciable change in core reactivity (Table 10-7). The use of non-hardened cross sections predicts no appreciable change in core reactivity but results in a higher innercell peaking over the presently used hardened values; however, this fact alone is not enough to justify the use of non-hardened cross sections since some hardening is certainly expected due to the preferential absorption of low energy neutrons.

It is possible that there is a significant change in spectrum throughout the element. This has not been included in any of the methods studied here; therefore, until a more detailed analysis of the thermal treatment of APPR-type fuel elements is made, the presently used model seems adequate and offers a good compromise between the accurate prediction of innercell thermal flux distributions and calculated core reactivity.

This work, including the theory and equations utilized, will be issued as an AP Note.*

* Bobe, P. E., "Effect of Neutron Temperature and Computational Model Upon the Thermal Constants of the SM-2 Core," AP Note to be issued.

TABLE 10-7
NUCLEAR PARAMETERS OF SM-2 FIXED ELEMENTS
VS. CALCULATIONAL MODEL

(0 Burnup, T 68°F, P 2000 psia)

Parameter	P ₃ (M.B.) Hardened *	P ₃ (W-W) **	P ₁ (M.B.) Hardened +	P ₃ (M.B.) Non-Hardened ++
$\sum aT$ (cm ⁻¹)	0.420882	0.309712	0.427266	0.471971
$\nu \sum fT$ (cm ⁻¹)	0.613246	0.450926	0.624381	0.690359
K _{th}	1.457050	1.455953	1.461340	1.462714
$\sum sT$ (cm ⁻¹)	2.036880	2.006063	2.038132	2.273421
D _T (cm)	0.160880	0.196249	0.160329	0.138107
γ (cm ²)	31.694438	31.694438	31.694438	31.694438
P	0.611830	0.611830	0.611830	0.611830
D _F (cm)	1.276164	1.276164	1.276164	1.276164
$\nu \sum ff$ (cm ⁻¹)	0.019972	0.019972	0.019972	0.019972
K _F	1.277800	1.277800	1.277800	1.277800
B ² (cm ⁻²)	0.006881	0.006881	0.006881	0.006881
K _{eff}	1.137135	1.135329	1.139318	1.140423
ρ (%)	12.06	11.92	12.23	12.39

* P₃ theory with cross sections averaged over a Maxwell-Boltzmann at a hardened neutron temperature (E = 0.0331 ev).

** P₃ theory, cross sections averaged over a Wigner-Wilkins spectrum

+ P₁ theory, cross sections averaged over a Maxwell-Boltzmann at a hardened neutron temperature (E = 0.0331 ev).

++ P₃ theory, cross sections averaged over a Maxwell-Boltzmann at a non-hardened neutron temperature (E = 0.0253 ev).

A-10

3. Calculation of Local Power Peaking Factors from Innercell Thermal Flux Distributions

When the P_3 theory model is employed to calculate the innercell thermal flux distributions (Section 2h of this report), the fast flux distribution is not available since there is a large fraction of epithermal fissions. In order to estimate the local power peaking, the power distribution is calculated by assuming a flat, fast flux distribution across the element. If the thermal flux within an element is normalized to the average flux within the active (fuel bearing) region, the max. to average power ratio is calculated by:

$$P_{\max}/P_{\text{avg}} = \bar{\beta} \phi_{\text{th max}} + (1 - \bar{\beta})$$

where $\bar{\beta}$ is the average percent of thermal fissions.

To test the validity and determine the effect of the fast flux distribution, of this expression, a multiregion, three group, one-dimensional VALPROD was calculated through the SM-2 core parallel to the fuel plates. The percent of thermal fissions, β , was calculated as a function of position within a fixed fuel element. The average value as calculated by these flux distributions, β_{flux} , was found to be 0.64773. The value of β is normally calculated by a knowledge of the fast and thermal parameters of the core, such that

$$\beta = \frac{P K_{\text{th}}}{K_{\text{eff}} (1 + \gamma B^2) (1 + L^2 B^2)} = \bar{\beta}_{\text{element}}$$

This β , called $\bar{\beta}_{\text{element}}$, is calculated to be 0.6419.

Another possible β to be used, in order to predict the calculated power distribution, may be calculated by a knowledge of the calculated thermal flux and calculated power peaking within the element. This value, called β_{cal} was found to be 0.654878. The max. to average power ratio as calculated by the various β 's are:

$$P_{\max}/P_{\text{avg flux}} = 1.359$$

$$P_{\max}/P_{\text{avg element}} = 1.356$$

$$P_{\max}/P_{\text{avg cal}} = 1.363$$

In general, as seen in Section 3 of this report, (Task 10.2) the calculated power peaking is considerably less than the experimentally measured values. Therefore, a correction factor is needed to correct calculated distributions; however, no additional factor is necessary to allow for errors in the selection of β since the maximum change in $P_{\max}/P_{\text{avg}}$, as calculated by the three methods discussed in this section, is only 0.52%. This error is well within the nuclear uncertainty factor of 5% to 10% normally used in reporting power distributions.

3. Analysis of Critical Experiments (Task 10.2) -B.E. Fried, P.E. Bobe

a. Analytical Reactivity Calculations

Reactivity calculations on the SM-2 Experimental Core were made using three different machine codes, namely:

1. The PDQ code, a two-dimensional code (X, Y or R, Z) for the IBM-704 machine, used for (X, Y) calculations.
2. The VALPROD code, a one-dimensional code for the IBM-650 machine, used for radial calculations (cylindrical geometry).
3. The FINK code, * a two group, two region code for the IBM-650 which solves for reactivity and the radial and axial reflector savings by iterating between radial and axial calculations, assuming a cylindrical core.

The equivalence of these calculations was demonstrated by computing the same core with all three codes (Table 10-8). The excellent agreement between these calculations shows that the assumption, that the 7 x 7 array with no corner elements can be calculated as a cylindrical core, is good for reactivity calculations. In all other comparisons made, the difference between the one dimensional radial calculations and the two dimensional, X Y, calculation was less than 0.2%. *f*

* APAE Memo No. 121, "Fink I: a Two Group, Two Region Calculation on the IBM-650," by P. E. McElligott, Issued January 28, 1958.

TABLE 10-8
EQUIVALENCE OF ANALYTICAL CODES

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

Code	K _{eff}	ρ %
PDQ-XY	1.141	12.4
VALPROD	1.140	12.3
FINK	1.140	12.3

b. Reactivity of the SM-2 Experimental Core

The reactivity of the SM-2 Experimental Core was calculated for various configurations (Table 10-9). The reference core (stainless steel reflector) reactivity at 68°F is 12.0% ρ with seven control rod fuel elements. This is a gain of 0.7% ρ over the reactivity of the SM-2 core at 68°F with a water reflector. The experimental value of the reactivity of the reference core is 11.5% ρ . The gain in reactivity using a large steel reflector instead of water was experimentally extrapolated to 0.6%.

The effect of substituting fixed fuel elements in place of control rod fuel elements was calculated as + 1.0% ρ . Measured values were obtained by substituting for one C.R.F.E. at a time, the total effect being 2.0% ρ . The total effect was obtained by adding up the individual effects. This measured value may be lower if all seven elements could be replaced at one time, since there are "interaction effects" on the flux when just one element is replaced. In this case, the core array becomes non-symmetrical and there is a shifting of the flux to the side containing the additional fixed fuel element. Adding the effects of each individual element tends to overestimate the effect of replacing all seven control rod fuel elements.

The experimental reactivities must be corrected for the use of boron in mylar rather than dispersed (in rather large lumps) in the matrix. This correction has been estimated analytically to be a maximum of 0.75% ρ . * This will be checked experimentally using substitution plates.

* APAE Memo No. 215, SM-2 Core and Vessel Monthly Report
May 28, 1959 to July 9, 1959 (Figure 8-2)

TABLE 10-9
REACTIVITY OF VARIOUS CONFIGURATIONS OF THE
SM-2 EXPERIMENTAL CORE - 68°F

Core Configuration	Percent Reactivity (% ρ) Calculated	Measured (% ρ)
Seven Control Rod Fuel Elements, SS Reflector	12.0	11.5*
All Fixed Fuel Elements, SS Reflector	13.0	13.5**
7 CRFE, Water Reflector	11.3	10.9
All F.F.E., Water Reflector	12.3	12.9***

* This value obtained using total K_{excess} in cents and $\rho = 1 - e^{-K_{\text{excess}}} \times 0.0078$.

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

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

The reactivity change in replacing the central control rod fuel element alone was calculated as + 0.2% ρ and measured as + 0.3% ρ , which is in much better agreement than that calculated for seven control rod elements.

c. Reactivity of the SM-2 Core at 510°F

The reactivity of the reference SM-2 core at operating temperature (510°F) with a stainless steel reflector and no xenon was calculated as 5.1% ρ . (Table 10-10).

The stainless steel reflector is now calculated to be worth 0.8% more than a water reflector. The effect of substituting fixed fuel elements for the control rod fuel elements was calculated to be +1.3%.

The loss of reactivity from 68°F to 510°F is 6.9% ρ for the SM-2 core with a stainless steel reflector.

At present, the best estimate of the reactivities of the SM-2 core is given in Table 10-11.

TABLE 10-10

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

Core Configuration	Calculated Reactivity (% ρ)
Seven Control Rod Fuel Elements, SS Reflector	5.1
7 C.R.F.E, Water Reflector	4.3
All Fixed Fuel Elements, SS Reflector	6.4
All F.F.E., Water Reflector	5.5

TABLE 10-11

BEST ESTIMATE OF SM-2 REFERENCE CORE REACTIVITY

Condition	Reactivity - %	
	Calculated	Measured
68°F, Clean, 0% Burnup	11.3*	11.5+
510°F, Clean, 0% Burnup, No Xenon	5.1*	4.8++
510°F, Eq. Xe, 0% Burnup	2.6**	2.3**
510°F, Max. Xe, 0% Burnup	0.8**	0.5**

* Calculated by PDQ (IBM-704 Code), corrected for flow divider loss of 0.7% ρ or 96.6 cents.

** From Section 2a, the zero to equilibrium and zero to peak xenon reactivity change are 2.5% and 4.3% respectively.

+ Taken from integral worth curve of 7 rod bank, corrected for flow divider loss of 0.7% ρ or 96.6 cents.

++ The measured hot to cold change is 6.7 ρ or 890 cents.

d. Revised Flux and Power Distributions

Since the PDQ code was made available to Alco during the present report period, revised flux and power distributions for both the SM-2 Experimental Core at 68⁰F and the SM-2 Reference Core at 510⁰F have been made.

Comparisons with PDQ and previous one-dimensional calculations on the Valprod code show that the analytical model used to predict gross power distributions over most of the core results in remarkable agreement.* (See Table 10-12). The extrapolations of the one-dimensional calculations in APAE Memo 195 predict the shape and magnitude of the power over the core as accurately as the PDQ two-dimensional calculation.

Figure 10-9 shows a comparison of a one-dimensional slab calculation and a PDQ (X, Y) calculation through the core central plane. The two curves are identical when normalized to the same value at the core centerline. However, the centerline values are considerably different. If the slab Valprod calculation is corrected by the centerline value of the radial Valprod calculation, which is accurate, then good agreement is obtained.

* Byrne, B. J., "Power Distribution for APPR-Type Cores," APAE Memo 195, Issued July 28, 1959

TABLE 10-12

COMPARISON OF POWER DISTRIBUTIONS AS PREDICTED
BY THE PDQ CODE AND ANALYTICAL
EXTRAPOLATIONS OF THE VALPROD CODE

(SM-2 Experimental Core - 68°F - All Fixed Fuel Elements,
Homogenized Properties - Water Reflector)

Code	$\frac{P \text{ core center}}{P \text{ core avg of 1.0}}$	$\frac{P \text{ core edge}}{P \text{ core avg of 1.0}}$	$\frac{P \text{ edge}}{P \text{ center}}$
PDQ - through central plane of core	1.56	1.71	1.10
PDQ - through plane where the distance from the center of the core to the edge of the core is equal to the equivalent radius of the cylindrical core	1.56	1.44	0.92
Radial VALPROD	1.58	1.51	0.96
Radial Valprod- Extrapolated to the central plane of the core as in APAE Memo No. 195	1.58	1.74	1.10
SLAB VALPROD - through central plane of the core	1.28	1.41	1.10

Intercell thermal flux distributions which have been obtained from the P-3 code were compared to recently obtained thermal flux distributions from PDQ (x, y) calculations (P-1 or diffusion theory calculations). As seen in Figures 10-10 and 10-11, use of PDQ predicts a higher peak at the edge of the active (fuel) region in the 68°F core. At 510°F, both calculations are in good agreement and show a decrease in flux peaking with an increase in temperature.

Figure 10-12 shows a two-dimensional power contour of the SM-2 Experimental Core at 68°F with a stainless steel reflector. Local power peaking in the elements is included. The worst power peak in the core occurs in the corners of the fixed fuel element adjacent to the central control rod fuel element.

3. Comparison of Calculated and Measured Power Distributions

Measurements of radial and axial power distributions in the SM-2 Experimental Core at 68°F were taken with uranium and gold foils.

Figures 10-13 and 10-14 show comparisons between calculated (PDQ-XY) and experimental radial distributions for both water and steel reflected cores. The agreement is good except at the edge of the elements, where the calculated value underestimates the power peak considerably.

Figures 10-15, 10-16, 10-17 and 10-18 show the comparison between calculated and measured axial power distributions. The analytical values were obtained from the Windowshade code. The homogeneous bank position was at 7.95" out while the measured bank position was 6.97" out. This is a difference of 0.98 inches. In SM-1, the difference between the Windowshade bank and the measured bank position was 0.90 inches. The agreement between the measured and calculated axial power distributions seems quite good over a wide range of fixed fuel element positions. The experimental peak to average power is lower than the calculated value since most of the measured points fall above the calculated curve. Each of the experimental axial traverses was taken down the center of the element.

In order to obtain the power generation for the hot spot and worst plate in each element, $Q(\Delta\theta)$ and $Q(\Delta T)$ respectively, it was necessary to first obtain correction factors to bring the calculated values into agreement with the measured power distributions.

The complete expression for the radial hot spot factor at the start of life is:

$$Q(\Delta\theta) = P(X, Y)_{\max} P(Z) F_{\text{rods}} F_{\text{local}} \text{correction}$$

where: $Q(\Delta\theta) =$ Power generation rate at the hottest spot in each element, normalized to an average radial power of 1 over the core.

$$F_{\text{rods}} = \frac{\left(\frac{P(X, Y, Z) \text{ measured at the center of each element}}{P \text{ measured core avg. of 1.0}} \right)}{\left(\frac{P(X, Y)P(Z) \text{ calculated at the center of each element}}{P \text{ calculated core avg. of 1.0}} \right)}$$

$$F_{\text{local correction}} = \frac{\left(\frac{P(X, Y, Z) \text{ measured at the edge of each element}}{P(X, Y, Z) \text{ measured at the center of each element}} \right)}{\left(\frac{P(X, Y) \text{ calculated at the edge of each element}}{P(X, Y) \text{ calculated at the center of each element}} \right)}$$

$P(X, Y)_{\text{max}} P(Z)_{\text{max}}$ = maximum calculated power generation in each element

The local peaking correction factor, $F_{\text{local correction}}$, was calculated at the edge of the element instead of the corner simply because no measurements were taken at corner points. The correction factors, on the calculated power, F_{rods} and $F_{\text{local correction}}$, were assumed to remain constant with temperature. Since the worst hot spot occurs near core midlife when the rods are furthest in, the axial power distribution was taken at this point. However, the radial power peak in the corner of the elements has been decreased since the fuel has been depleted at a faster rate. This correction, calculated at the edge of the elements, has also been applied to the hot spot factor at the corner of the elements.

To obtain the power generation rate in the hottest plate in each element, $Q(\Delta T)$, the factor $Q(\Delta \theta)$ was multiplied by the ratio of the average power in the hottest plate to the power at the hottest spot in each element. (See Table 10-12). The radial power distributions in Table 10-12 must be multiplied by the worst maximum to average axial power distribution in order to obtain the overall peaking factors. These axial peaking factors were obtained from the axial burnout calculations at 440 F previously reported,* and are reproduced in Table 10-13.

The recommended nuclear uncertainty factor, $F_n(\Delta \theta)$, has been reduced from 1.10 to 1.05, since local peaking factors have been predicted more accurately by the PDQ-X, Y code and corrected to agree with the

*APAE Memo No. 223, SM-2 Reactor Core and Vessel Review Report, May 28, 1959 to August 24, 1959 (Table 8-8).

measurements. Some conservatism has been introduced in the local burnup correction since the burnup at the element edge was used. The burnup factor in the corner of the element will be smaller due to greater fuel depletion.

TABLE 10-12

SM-2 RADIAL POWER DISTRIBUTION

SM-2 Reference Core - 510⁰F - S.S. Reflector - Midlife)

Element No.	Q (ΔT)	Q ($\Delta \theta$)
44	1.66	1.95
45	1.62	1.88
46	1.19	1.57
47	1.15	1.37
34	1.41	1.93
35	1.37	1.93
37	1.08	1.35
25	1.37	1.80
26	1.16	1.57
27	0.71	1.01
14	0.89	1.51
15	0.88	1.49
16	0.62	1.07

$$P(Z)_{\max.} = 1.84$$

$$F_N(\Delta T) = 1.05$$

$$F_N(\Delta \theta) = 1.05$$

$$\psi(\Delta T) = 1.00$$

$$\psi(\Delta \theta) = 0.95$$

where:

Q (ΔT) = power generation for worst channel in each element, normalized to a core average of 1.0.

Q ($\Delta \theta$) = power generation at the hottest spot in each element, normalized to a core average of 1.0.

P (Z)_{max.} = power generation at worst axial point in the core, normalized to a core average of 1.0.

F_N(ΔT) = nuclear hot spot factors to allow for uncertainty in the

F_N($\Delta \theta$) = calculated power distributions.

$\psi(\Delta T)$ = fraction of total reactor power generated in the core.

$\psi(\Delta \theta)$ = fraction of total reactor power generated in the fuel plates.

TABLE 10-13

**CORRECTED WINDOWSHADE (AXIAL) POWER DISTRIBUTION
AT THE DEEPEST ROD BANK INSERTION
- (440°F - NO XENON)**

Inches From Bottom of Core	P(Z)/P core average of 1.0
0	2.92 (Bottom Spike)
2	1.56
4	1.80
5.5	1.84 (Axial Max.)
6	1.83
8	1.55
10	0.97
12	0.74
14	0.58
16	0.43
18	0.31
20	0.20
22	0.17

4. Development of Analytical Techniques (Task 10.3)-R. L. Murray

a. Calculation of Effective Delayed Neutron Fraction

The value of the effective delayed neutron fraction (β) as calculated and reported in the last review report (APAE Memo 223) has been revised. The value was calculated by a theory developed by R. L. Murray.* This theory derives an analytical expression for β using the modified two-group theory of a reflected spherical core. Using the nuclear parameters of the core presented in APAE Memo 197, the value of the effective β was found to be 0.00777 instead of the previously calculated value of 0.00724. This value may be compared to the value of 0.0073 which has been used in the interpretation of the SM-2 Flexible Critical Experiments.

b. Radial Ring Equivalents of Local Perturbations

A method has been developed to determine appropriate radial ring equivalents of local perturbations, such as the presence of control rod follower fuel elements. The effect of radial zones may then be computed by the VALPROD and NUB (non uniform burnup) codes. Four criteria

* Murray, R. L., "Effective Delayed Neutron Fraction," AP Note 171, August 18, 1959.

for equivalence of the sum of all local regions to two rings are established - equal volume of materials, equal power contribution, equal initial reactivity worth, and equal reactivity associated with fuel consumption, to first order. The mathematical formulation consists of four simultaneous equations.

$$\sum_i v_i (\bar{\phi})_i^n = \sum_j v_j (\bar{\phi})_j^n$$

where i refers to the several local regions, j to the rings, and there appears averages over the volumes of the thermal fluxes to various powers n. Analytic and tabular values of the averages have been derived. An AP Note giving this information is in preparation.

FIG. 10-1

FIG. 10-2

FIG. 10-3

FIG. 10-4

GAMMA HEAT GENERATION
ON THE INNER SURFACE OF
THE PRESSURE VESSEL.

FIG. 10-5

RELATIVE FLUX PARALLEL TO FUEL PLATES

FIG. 10-6

RELATIVE FLUX PARALLEL TO FUEL PLATES

FIG. 10-7

FIG. 10-8

RELATIVE POWER DISTRIBUTION THROUGH ONE SIDE
OF THE SM-2 EXPERIMENTAL CORE - 68°F

FIG. 10-9

FIG. 10-10

THERMAL FLUX DISTRIBUTION IN A FIXED ELEMENT
ADJACENT TO A FIXED ELEMENT IN THE SM-2 CORE - 510°F

FIG. 10-11

Fig. 10-12 SM-2 POWER CONTOUR
68°F
STAINLESS STEEL REFLECTOR

FIG. 10-13

022 048

FIG. 10-14

FIG. 10-15

FIG. 10-16

FIG. 10-17

FIG. 10-18

B. CORE THERMAL DESIGN

1. Progress (Task 2.0) - I. Beretsky

The SM-2 final flow evaluation was made using a set of hot channel factors based on the latest set of reactor manufacturing tolerances. The new total required flow in the reactor is 7800 gpm. This new flow rate is 91% of the previous reported value of 8560 gpm*. This reduction of total flow requirements was brought about by the increased heat transfer area and reduced maximum heat flux.

All other hot channel factors were calculated using APAE Memo 157** as reference.

2. Deflection of Outer Fuel Plates Adjacent to Lattice Region (Task 2.8) - I. Beretsky

If we consider the lattice passage area ($b \times w$) to be deflected inwards at the midpoint a distance δ as shown below,

* APAE Memo 223, SM-2 Reactor and Vessel Review Report, May 28, 1959 to August 24, 1959.

** APAE Memo 157, "Thermal Design Criteria for APPR Pressurized Water Reactors," J.D. Love

Then the reduction in area can be represented by

$$\text{Area Reduction} = bw - 2 \int_0^w y dx = A_r$$

$$\text{If } y \text{ is assumed} = \delta \sin \frac{\pi x}{w}$$

$$\text{Then } A_r = bw - 2 \delta \int_0^w \sin \frac{\pi x}{w} dx = bw - \frac{4 \delta w}{\pi}$$

$$\text{Or } \frac{A_{\text{Final}}}{A_{\text{Initial}}} = 1 - \frac{4 \delta}{b \pi} = \gamma$$

However,

$$T_s = T_{\text{inlet}} + \Delta T_w + \Delta T_f$$

where T_s = Surface temperature, $^{\circ}\text{F}$

T_{inlet} = Inlet temp, $^{\circ}\text{F}$

ΔT_w = Water temperature, Rise, $^{\circ}\text{F}$

ΔT_f = Film temperature drop, $^{\circ}\text{F}$

Therefore, a reduction in flow area will tend to increase both ΔT_w and ΔT_f . In order to compensate for this effect, the flow must be increased in order to maintain the same ΔT_w and ΔT_f such that the surface temperature remains the same.

$$\text{Since } \Delta T_w \propto \frac{1}{W} \text{ and } W \propto A$$

$$\frac{T_{\text{final}}}{T_{\text{initial}}} = \frac{W_{\text{initial}}}{W_{\text{final}}} = \frac{A_{\text{initial}}}{A_{\text{final}}} = 1$$

$$\text{Since } \frac{A_{\text{initial}}}{A_{\text{final}}} = \frac{1}{\gamma}$$

$$\frac{W_{\text{final}}}{W_{\text{initial}}} = \frac{W_{\text{req'd}}}{W_{\text{calculated}}} = \frac{1}{\gamma} = \frac{1}{1 - \frac{4 \delta}{b \pi}} = R$$

$$\text{or } R = \sum_{n=0}^{\infty} \left(\frac{4 \delta}{b \pi} \right)^n = 1 + \left(\frac{4 \delta}{b \pi} \right) + \dots + \left(\frac{4 \delta}{b \pi} \right)^n + \dots$$

$$\text{where } \delta = \frac{\delta}{b}$$

For a small deflection

$$R \approx 1 + \frac{4\zeta}{\pi} + \frac{16\zeta^2}{\pi^2}$$

The values used to calculate R were δ equal 0.002" and b equal 0.123".

3. Final Flow Evaluation (Task 2.8) - I. Beretsky, L.S. Hecht

Since the last Review Report (APAE Memo 223) a number of changes in parameters affecting SM-2 volume flow requirements have occurred. The most important of these is an increase in fuel plate meat width, reduction in water gap tolerances and revision in most adverse power distribution.

Table 2-1 presents a summary of the factors considered in calculating hot channel factors. The nominal value, deviation, and average and local hot channel factors are presented for each item.

Table 2-2 presents a list of pertinent dimensions and data used in the SM-2 thermal analysis.

The latest adverse power distribution data is presented in Task 1.0. The method by which the volume flow requirements are established is given in APAE Memo 157.

An additional factor is included in Table 2-1, F_L , which is applied to determine how much more flow above that of the hottest fuel element channel, the lattice channel element must receive. This new factor is above the previously used value of 5% because of fuel plate deflections in the lattice channel (see Section 2) which was not considered in the previous analysis.

4. Flow Requirements - L.S. Hecht

Table 2-3 presents the required flow per element. The required volume flow thru the control rod assembly is 331 gpm compared to 363 gpm previously calculated. The highest required flow thru a fixed element in the second pass is 316 gpm compared to a previously calculated value of 384 gpm. The total required flow is 7203 gpm based on tailoring the fixed elements in the second pass. To facilitate rod programming in subsequent SM-2 cores it is desirable to have uniform flow thru each fixed element in the second pass. The total required flow in this case is 7446 gpm. A leakage allowance of 350 gpm should be included. Therefore the total reactor flow is 7800 gpm.

The reduction in volume flow requirement is due principally to the increased meat width and reduced maximum to average power generation.

TABLE 2-1
SUMMARY OF DIMENSIONAL DEVIATIONS AND HOT CHANNEL
FACTORS FOR SM-2 THERMAL DESIGN

Type of Deviation	Nominal Value	Deviation	F _{AVG}	F _{Local}
1. Plate Spacing	0.123"	+0.004" Avg +0.008" Local	1.0501	1.1062
2. Uranium Content	100%	+5%) -0.7%) Avg	-----	-----
3. Meat Length	22"	+0.5" Avg	1.0233	1.0233
4. Homogeneity	100%	+1% Local	-----	-----
5. Uranium Content (Including Homogeneity)	100%	(2) and (4) Combined	1.005	1.015
6. Clad Thickness	0.005"	+0.0003" Avg +0.0005" Local	1.0071	1.0119
7. Plate to Plate Maldistribution				
Fuel Element	100%	+6%	1.0638	1.0508
Control Rod	100%	+12%	1.1364	1.1080
9. Combined Factors				
Fuel Element	---	---	1.1569	1.2217
Control Rod	---	---	1.2315	1.2843
10. New Lattice Factor, F _L , F _L = F _L ' x R	$\delta = 0.002"$ $b = 0.123"$	F _L ' = 1.0435 (old lattice factor) R = 1.0211 (deflection factor)		

TABLE 2-2
DATA USED IN SM-2 FLOW DETERMINATION

<u>Symbol*</u>	<u>Definition</u>	<u>Value</u>
A_{HT}	Heat Transfer Area, ft^2	587.7
f_{p1}	Fraction of Power Generated in 1st Pass	0.4048
T_{in1}	Inlet Temperature to the 1st Pass, $^{\circ}F$	500
T_{in2}	Inlet Temperature to the 2nd Pass, $^{\circ}F$	512
ΔX	Clad Thickness, in	0.005
ΔY	Meat Thickness, in	0.030
ϕ	Average Heat Flux, $BTU\ Hr^{-1}Ft^{-2}$	168,300
I, J	Dimensions of Unit Cell, in	2.9375
V, Y	Inside Control Rod Basket Dimensions, in.	2.7255
K, U	Outside Control Rod Basket Dimensions, in	2.8255
B_{FE}	Width of Side Plate, in	2.863
B_{CR}	Width of Side Plate, in	2.619
C_T	Fuel Plate Thickness, in	0.040
E	Side Plate Thickness, in	0.040
S_{FE}	Width of Element Water Gap, in	2.789
S_{CR}	Width of Element Water Gap, in	2.533
t	Thickness of Element Water Gap, in	0.123
M_{FE}	Active Meat Width of a Fuel Plate, in	2.670
M_{CR}	Active Meat Width of a Fuel Plate, in	2.363
H_{FE}	Active Meat Length of a Fuel Plate, in	22.0
H_{CR}	Active Meat Length of a Fuel Plate, in	21.5
i_{CR}	Meat Length of Control Rod Fuel Plates inserted in the Active Core, in	7.6

* APAE Memo 161, "Lovely-I, Program No. 55, Thermal Analysis on the IBM-650," J.D. Love and C. Stueck.

TABLE 2-3
REQUIRED FLOW AND THERMAL DATA FOR THE SM-2

Required Flow Per Element

<u>Pass No.</u>	<u>Element No.</u>	<u>No. of Similar Elements</u>	<u>Ref. Case (APAE Memo 223)</u>	<u>Final Flow Requirements</u>
1	14	2	310	202
1	15	4	306	199
1	16	4	242	133
1	26	4	290	218
1	27	4	---	127
1	37	4	---	184
1	47	2	---	189
2	25	4	350	292
2	34	2	384	316
2	35	4	333	315
2	42	2	342	248
2	44 (control rod)	7	363	331
2	45	2	---	313

Total Flow and Thermal Data

		<u>Previous Ref. Case (AP Memo 223)</u>	<u>Final Requirements</u>
Internal	1st Pass	6789	4226
Flow, GPM	2nd Pass	7494	6498
Lattice	1st Pass	878	526
Flow, GPM	2nd Pass	1069	705
*Total Req'd	1st Pass	7667	4752
Flow, GPM	2nd Pass	8560	7203
Reflector			
Flow, GPM		893	2450
Maximum Internal			
Temp. °F		704	697
Total Flow, GPM		8560	7203
*Uniform Total	Pass 1	8320	5758
Req'd Flow	Pass 2	8990	7446

* Excluding Leakage flow between and across passes.

C. REACTOR VESSEL DESIGN

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

The type 304 stainless steel vessel drawing has been prepared and sent out for cost estimates.

Tentative designs of the core support and fuel elements structure have been made and are being evaluated for pressure drop and flow distribution.

Drawings of the full scale closure have been completed and sent out for bids.

Stress analysis on several bolting materials have been made. In order to meet the ASME Code, a selection of 24 three-inch type 403 stainless steel bolts appears to be the best.

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

A tentative design of the stationary fuel element has been made and is shown on Dwg. R9-13-1017. The problems confronting the designer are: (1) how to keep the fuel element length as short as possible; (2) how to provide the best possible distribution within the element; (3) how to raise the pressure drop through the fixed element to that across the control rod assembly; (4) how to permit easy handling of the element.

Recently it became necessary to add a comb to the entrance end of the fuel plates to assure no collapse of the flow channels. As presently designed the comb gives the element some structural stability, provides a means for locating the element in the core, and facilitates handling. All the fuel plates have been made the same length for ease of fabrication and to help eliminate the rippling problem in the outer plates. This was suggested by ORNL metallurgy.

Very recently it became necessary to add a flux suppressor to the lower end of the fuel element to get rid of the power spike. The active length of the element has been kept the same; therefore the fuel plate is slightly longer to include the suppressor.

Three types of end fixtures are shown on the fuel element drawing. The casting (Style C) would be the most attractive from the mechanical design standpoint because it presents a stronger fuel element, and the machining costs should be lower than the other types.

The other types (Style A and B) are possible methods which can be fabricated from plate stock and welded together. Either will be satisfactory as far as design is concerned. The economics and fabrication problems will dictate which is the better design.

The dimensions for the fuel as shown on this drawing were supplied by personnel on other tasks.

The control rod fuel element is shown on Dwg. C9-13-1028, Rev. A. The outside dimensions of this element are the same as SM-1. The two long fuel plates at the lower end are used to orientate the element in the control rod tube and to prevent the control rod from being incorrectly loaded. The two long plates at the top eliminate the separate handle of previous designs. The elimination of the handle permitted the absorber to be brought slightly closer to the fuel element to help with the peaking problem. The drawing provides for assembly by TIG welding.

The control rod assembly drawing AEL 445 (APAE Memo 197) is still the reference. The cross-section of the tube has been reduced to make it the same as SM-1. Since europium is being used as the absorber material it is no longer felt necessary to allow for growth of the absorber. The smaller size tube keeps the outer passages in the fuel element the same as the internal passages.

The fuel element spacing reference drawing is AES 319 (APAE Memo 197).

3. Core Support Structure (Task 3.3) - T. F. Connolly

A design drawing of the core support structure is shown on AEL 475. Cost estimates on the core support structure are presently being prepared.

A design of the bottom plate (AEL 523) was prepared and the first hydraulic test results were favorable. This plate is 1-1/2" thick with many small diameter holes to provide the best flow distribution both within the fuel element and in the lattice passages. The center hole for each fuel element must be accurately sized and spaced, but tolerances on the rest may be somewhat relaxed. Slots on the top side orientate the fuel elements and the holes are provided for location. There are also slots in the top to orientate and contain the flow divider.

The first design of the top doors of the core support structure is shown on AEL 526. Doors were tentatively selected because the lifting point could be on the centerline of the door, thus eliminating the unbalanced conditions as noted in SM-1. There is no storage problem when doors are used.

The first test results of these doors were not favorable (See Task 6); and further work will be necessary. A second design will be prepared using holes similar to those used in the bottom plate, which should improve the pressure drop conditions across the doors.

4. Vessel Design (Task 3.5) - T. F. Connolly

The latest vessel design is shown on AEL 521. The main difference of this over previous designs is the change to type 304 stainless steel, which increases the wall thickness.

Calculations have been made on the thru bolts using different materials for bolts and nuts. To stay within the ASME Code, type 403 stainless steel is the best material available. A back-up material A-286 has also been used in the calculations, but this is not a code approved material. Previous calculations on stud and bolt materials have been based on 20,000 psi stress values, but in the case of type 403 it was found necessary to drop this to 17,850 psi; therefore, 24 three-inch studs will be necessary to maintain an effective seal. Also, the coefficients of expansion of type 403 and 304 are different, which will put some added stresses into the bolts at operating conditions.

5. Closure Development (Task 3.9) - T. F. Connolly

The design of the full scale closure rig (AEL-519) has been completed and sent out for bids. The same problem of type 403 bolts and type 304 vessel exists here as in the above discussion. Therefore, some change will be necessary before fabrication starts. In the full scale closure test, the actual stresses due to pressure and temperature will be determined.

A suggested test set-up is shown on AEL 525. As shown, pressurizing is by means of an internal heater. Consideration is being given to a pressure pump or bottled gas. The points for recording temperature are tentative. Temperatures of the vessel flange (internal and external), gasket (internal and external), cover (internal and external), and bolts will be recorded. Measurements of bolt length will be taken before, during, and after all tests. Galling tests of several studs (both 403 and A286) will be conducted. Corrosion tests on vessel, gasket, and bolting material will be run.

6. Hydraulic Balanced Control Rod - J. F. Haines

A design study for a hydraulic balanced control rod has been carried out. This approach minimizes the effect of the pressure drop across the

control rod. It might be necessary to incorporate this approach if further increases in pressure drop across the control rod are required. The design is feasible but adds complication to the reactor.

In order to get the transfer tubes from the bottom to top chambers it will be necessary to cut into the outer reflector ring. Studies would be necessary to determine the effect of these slots on vessel thermal heating and ring stability. Studies might also be made of the feasibility of running the tubes external to the vessel.

A development program would be necessary to determine the balance necessary for proper control rod operation, the effect of leakage on the pump size, and life testing of the seals employed.

DATE	1/2	REVIEW	RECORD	-	AUTH	CH. 200
INT-57	A	SIDE PLATE SLOT DRILL				
		ADDED. BASES TURNED				
		70° AND SUPPORTS ROOF				
		COVER ADDED & WELDING				
		METHOD CHANGED.			JK	OK
1/2-57	B	FUEL CLEANER ENDS				
		CHANGED. DRY. 2.12.6				
		WMS 1. 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. 101. 102. 103. 104. 105. 106. 107. 108. 109. 110. 111. 112. 113. 114. 115. 116. 117. 118. 119. 120. 121. 122. 123. 124. 125. 126. 127. 128. 129. 130. 131. 132. 133. 134. 135. 136. 137. 138. 139. 140. 141. 142. 143. 144. 145. 146. 147. 148. 149. 150. 151. 152. 153. 154. 155. 156. 157. 158. 159. 160. 161. 162. 163. 164. 165. 166. 167. 168. 169. 170. 171. 172. 173. 174. 175. 176. 177. 178. 179. 180. 181. 182. 183. 184. 185. 186. 187. 188. 189. 190. 191. 192. 193. 194. 195. 196. 197. 198. 199. 200. 201. 202. 203. 204. 205. 206. 207. 208. 209. 210. 211. 212. 213. 214. 215. 216. 217. 218. 219. 220. 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1017. 1018. 1019. 1020. 1021. 1022. 1023. 1024. 1025. 1026. 1027. 1028. 1029. 1030. 1031. 1032. 1033. 1034. 1035. 1036. 1037. 1038. 1039. 1040. 1041. 1042. 1043. 1044. 1045. 1046. 1047. 1048. 1049. 1050. 1051. 1052. 1053. 1054. 1055. 1056. 1057. 1058. 1059. 1060. 1061. 1062. 1063. 1064. 1065. 1066. 1067. 1068. 1069. 1070. 1071. 1072. 1073. 1074. 1075. 1076. 1077. 1078. 1079. 1080. 1081. 1082. 1083. 1084. 1085. 1086. 1087. 1088. 1089. 1090. 1091. 1092. 1093. 1094. 1095. 1096. 1097. 1098. 1099. 1100. 1101. 1102. 1103. 1104. 1105. 1106. 1107. 1108. 1109. 1110. 1111. 1112. 1113. 1114. 1115. 1116. 1117. 1118. 1119. 1120. 1121. 1122. 1123. 1124. 1125. 1126. 1127. 1128. 1129. 1130. 1131. 1132. 1133. 1134. 1135. 1136. 1137. 1138. 1139. 1140. 1141. 1142. 1143. 1144. 1145. 1146. 1147. 1148. 1149. 1150. 1151. 1152. 1153. 1154. 1155. 1156. 1157. 1158. 1159. 1160. 1161. 1162. 1163. 1164. 1165. 1166. 1167. 1168. 1169. 1170. 1171. 1172. 1173. 1174. 1175. 1176. 1177. 1178. 1179. 1180. 1181. 1182. 1183. 1184. 1185. 1186. 1187. 1188. 1189. 1190. 1191. 1192. 1193. 1194. 1195. 1196. 1197. 1198. 1199. 1200. 1201. 1202. 1203. 1204. 1205. 1206. 1207. 1208. 1209. 1210. 1211. 1212. 1213. 1214. 1215. 1216. 1217. 1218. 1219. 1220. 1221. 1222. 1223. 1224. 1225. 1226. 1227. 1228. 1229. 1230. 1231. 1232. 1233. 1234. 1235. 1236. 1237. 1238. 1239. 1240. 1241. 1242. 1243. 1244. 1245. 1246. 1247. 1248. 1249. 1250. 1251. 1252. 1253. 1254. 1255. 1256. 1257. 1258. 1259. 1260. 1261. 1262. 1263. 1264. 1265. 1266. 1267. 1268. 1269. 1270. 1271. 1272. 1273. 1274. 1275. 1276. 1277. 1278. 1279. 1280. 1281. 1282. 1283. 1284. 1285. 1286. 1287. 1288. 1289. 1290. 1291. 1292. 1293. 1294. 1295. 1296. 1297. 1298. 1299. 1300. 1301. 1302. 1303. 1304. 1305. 1306. 1307. 1308. 1309. 1310. 1311. 1312. 1313. 1314. 1315. 1316. 1317. 1318. 1319. 1320. 1321. 1322. 1323. 1324. 1325. 1326. 1327. 1328. 1329. 1330. 1331. 1332. 1333. 1334. 1335. 1336. 1337. 1338. 1339. 1340. 1341. 1342. 1343. 1344. 1345. 1346. 1347. 1348. 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1515. 1516. 1517. 1518. 1519. 1520. 1521. 1522. 1523. 1524. 1525. 1526. 1527. 1528. 1529. 1530. 1531. 1532. 1533. 1534. 1535. 1536. 1537. 1538. 1539. 1540. 1541. 1542. 1543. 1544. 1545. 1546. 1547. 1548. 1549. 1550. 1551. 1552. 1553. 1554. 1555. 1556. 1557. 1558. 1559. 1560. 1561. 1562. 1563. 1564. 1565. 1566. 1567. 1568. 1569. 1570. 1571. 1572. 1573. 1574. 1575. 1576. 1577. 1578. 1579. 1580. 1581. 1582. 1583. 1584. 1585. 1586. 1587. 1588. 1589. 1590. 1591. 1592. 1593. 1594. 1595. 1596. 1597. 1598. 1599. 1600. 1601. 1602. 1603. 1604. 1605. 1606. 1607. 1608. 1609. 1610. 1611. 1612. 1613. 1614. 1615. 1616. 1617. 1618. 1619. 1620. 1621. 1622. 1623. 1624. 1625. 1626. 1627. 1628. 1629. 1630. 1631. 1632. 1633. 1634. 1635. 1636. 1637. 1638. 1639. 1640. 1641. 1642. 1643. 1644. 1645. 1646. 1647. 1648. 1649. 1650. 1651. 1652. 1653. 1654. 1655. 1656. 1657. 1658. 1659. 1660. 1661. 1662. 1663. 1664. 1665. 1666. 1667. 1668. 1669. 1670. 1671. 1672. 1673. 1674. 1675. 1676. 1677. 1678. 1679. 1680. 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1847. 1848. 1849. 1850. 1851. 1852. 1853. 1854. 1855. 1856. 1857. 1858. 1859. 1860. 1861. 1862. 1863. 1864. 1865. 1866. 1867. 1868. 1869. 1870. 1871. 1872. 1873. 1874. 1875. 1876. 1877. 1878. 1879. 1880. 1881. 1882. 1883. 1884. 1885. 1886. 1887. 1888. 1889. 1890. 1891. 1892. 1893. 1894. 1895. 1896. 1897. 1898. 1899. 1900. 1901. 1902. 1903. 1904. 1905. 1906. 1907. 1908. 1909. 1910. 1911. 1912. 1913. 1914. 1915. 1916. 1917. 1918. 1919. 1920. 1921. 1922. 1923. 1924. 1925. 1926. 1927. 1928. 1929. 1930. 1931. 1932. 1933. 1934. 1935. 1936. 1937. 1938. 1939. 1940. 1941. 1942. 1943. 1944. 1945. 1946. 1947. 1948. 1949. 1950. 1951. 1952. 1953. 1954. 1955. 1956. 1957. 1958. 1959. 1960. 1961. 1962. 1963. 1964. 1965. 1966. 1967. 1968. 1969. 1970. 1971. 1972. 1973. 1974. 1975. 1976. 1977. 1978. 1979. 1980. 1981. 1982. 1983. 1984. 1985. 1986. 1987. 1988. 1989. 1990. 1991. 1992. 1993. 1994. 1995. 1996. 1997. 1998. 1999. 2000. 2001. 2002. 2003. 2004. 2005. 2006. 2007. 2008. 2009. 2010. 2011. 2012. 2013. 2014. 2015. 2016. 2017. 2018. 2019. 2020. 2021. 2022. 2023. 2024. 2025. 2026. 2027. 2028. 2029. 2030. 2031. 2032. 2033. 2034. 2035. 2036. 2037. 2038. 2039. 2040. 2041. 2042. 2043. 2044. 2045. 2046. 2047. 2048. 2049. 2050. 2051. 2052. 2053. 2054. 2055. 2056. 2057. 2058. 2059. 2060. 2061. 2062. 2063. 2064. 2065. 2066. 2067. 2068. 2069. 2070. 2071. 2072. 2073. 2074. 2075. 2076. 2077. 2078. 2079. 2080. 2081. 2082. 2083. 2084. 2085. 2086. 2087. 2088. 2089. 2090. 2091. 2092. 2093. 2094. 2095. 2096. 2097. 2098. 2099. 2100. 2101. 2102. 2103. 2104. 2105. 2106. 2107. 2108. 2109. 2110. 2111. 2112. 2113. 2114. 2115. 2116. 2117. 2118. 2119. 2120. 2121. 2122. 2123. 2124. 2125. 2126. 2127. 2128. 2129. 2130. 2131. 2132. 2133. 2134. 2135. 2136. 2137. 2138. 2139. 2140. 2141. 2142. 2143. 2144. 2145. 2146. 2147. 2148. 2149. 2150. 2151. 2152. 2153. 2154. 2155. 2156. 2157. 2158. 2159. 2160. 2161. 2162. 2163. 2164. 2165. 2166. 2167. 2168. 2169. 2170. 2171. 2172. 2173. 2174. 2175. 2176. 2177. 2178. 2179. 2180. 2181. 2182. 2183. 2184. 2185. 2186. 2187. 2

DATE	LET.	REVISION RECORD	AUTH.	DR.	CHK.
11-16-59	A	ENLARGED VIEW OF SIDE PLATE ADDED.			
		SECTION AT "C" ALTERED	JTC	HF	
12-4-59	B	DIMENSIONS ADDED FOR HANDLE.			
		CUTOUT REMOVED.	JTC	HF

(A) ENLARGED VIEW OF SIDE PLATE
2 X SIZE

ASSEMBLED ELEMENT MUST
FIT WITHIN A THEORETICAL
STRAIGHT BOX OF THESE
DIMENSIONS 25" LONG WITHOUT
INTERFERENCE

BREAK SHARP EDGES
REMOVE ALL BURRS

UNLESS OTHERWISE SPECIFIED
DIMENSIONS ARE IN INCHES.
TOLERANCES ON FINISHED
FRACTIONAL DIMENSIONS
TO BE ± 1/64

- (1) FINISH AS INDICATED
IN MICROINCHES.
- (1) MACHINE FINISH - ROUGH
- (10) FLAME CUT OR SAW

PRELIMINARY

ALCO		ALCO PRODUCTS, INC.	
NUCLEAR POWER ENGINEERING DEPT.		SCHENECTADY, N. Y., U. S. A.	
SCALE: FULL	REF.	DR.	RSC. 6/15/59
MATERIAL SPEC.		TR.	
		CHK.	
		APPR.	JTC 7/8/59
		APPR.	
		MET.	
NAME: FUEL ELEMENT (CONTROL ROD)			
PART NO. C-9-13-1028

SECTION A-A

LAYOUT			
ALCO PRODUCTS, INC. SCHENECTADY, NEW YORK ATOMIC ENERGY DEPT.			
MODEL - SM-2	STARTED 11-28-57	FINISHED 12-1-57	SCALE FOU L
REFERENCES -	DRAWN BY W. J. H. H.		
AUTHORITY -	DATE 11/28/57	APPROVED W. J. H. H.	DATE 11/28/57
	DATE	APPROVED	DATE
TITLE BOTTOM PLATE (CORE SUPPORT)			
LAYOUT NO. AEL-523

DATE	REVISION	REASON	BY	CHK
10/1/59	1	DESIGN	J. J. J.	
10/1/59	2	DESIGN	J. J. J.	
10/1/59	3	DESIGN	J. J. J.	
10/1/59	4	DESIGN	J. J. J.	
10/1/59	5	DESIGN	J. J. J.	
10/1/59	6	DESIGN	J. J. J.	
10/1/59	7	DESIGN	J. J. J.	
10/1/59	8	DESIGN	J. J. J.	
10/1/59	9	DESIGN	J. J. J.	
10/1/59	10	DESIGN	J. J. J.	
10/1/59	11	DESIGN	J. J. J.	
10/1/59	12	DESIGN	J. J. J.	
10/1/59	13	DESIGN	J. J. J.	
10/1/59	14	DESIGN	J. J. J.	
10/1/59	15	DESIGN	J. J. J.	
10/1/59	16	DESIGN	J. J. J.	
10/1/59	17	DESIGN	J. J. J.	
10/1/59	18	DESIGN	J. J. J.	
10/1/59	19	DESIGN	J. J. J.	
10/1/59	20	DESIGN	J. J. J.	
10/1/59	21	DESIGN	J. J. J.	
10/1/59	22	DESIGN	J. J. J.	
10/1/59	23	DESIGN	J. J. J.	
10/1/59	24	DESIGN	J. J. J.	
10/1/59	25	DESIGN	J. J. J.	
10/1/59	26	DESIGN	J. J. J.	
10/1/59	27	DESIGN	J. J. J.	
10/1/59	28	DESIGN	J. J. J.	
10/1/59	29	DESIGN	J. J. J.	
10/1/59	30	DESIGN	J. J. J.

ALTERNATE GASKET FORM
SCALE: 1/2 X SIZE

ENLARGED SECTION AT GASKET
SCALE: 1/2 X SIZE

1/2" - 8 BOLTS
16" - RECD.

DETAIL OF LIFTING LUG
2 RECD. R & L

3" - 8 THRU BOLTS
24 RECD.

LIP AROUND TOP OF COVER NOT
SHOWN IN THIS VIEW

POSSIBLE STUD ADDITION

4" BLIND FLANGE (2500)
2 RECD.

4" WELD NECK FLANGE (2500)
2 RECD.

FLEXITALL GASKET CG-25L
2 RECD.

4" NOZZLE
2 RECD.

MOUNTING PADS EQ. SP.
3 RECD.

HEATER WELLS
4 RECD.

EXPANSION JOINT (16 GA)

— MATERIAL —
VESSEL COVER, TOP FLANGE, SHELL & NOZZLES — FORGING SA336F8, PLATE SA240 TYPE 304
ELLIPTIC HEAD — SA336 F8
VESSEL SUPPORT, INSULATION JACKET, COVER LIFTING LUGS, SA240 TYPE 304
EXPANSION JOINT & SUPPORT & BOLT KEEPERS — SA240 GR S TYPE 304 PLATE.
OCTAGONAL GASKET — TYPE 304 OR B-160 GR A NICKLE (ANNEALED)
THRU BOLTS — TYPE 403 (CASE 1274N) OR AMS 5735 (A286)
NUTS, (THRU BOLTS) — TYPE 403 (CASE 1274N) OR AMS 5735 (A286)

— DESIGN NOTES —
DESIGN PRESSURE — 2200 P.S.I.
DESIGN TEMPERATURE — 600° F.
CORROSION ALLOWANCE — 0.1 IN.
HYDROSTATIC TEST PRESSURE — A.S.M.E. CODE.
CONSTRUCTION & WELDING IN ACCORDANCE WITH
LATEST A.S.M.E. CODE SECTION VIII CODE CASE 1234
AND ALCO PRODUCTS INC. SPEC. (LATER)

BOTTOM VIEW

LAYOUT	
ALCO PRODUCTS INC.	
ROULETTE, NEW YORK	
ATOMIC ENERGY DEPT.	
MODEL	SM-2
REFERENCE	AEL-469
AUTHORITY	
DATE	
APPROVED	
DATE	
APPROVED	
TITLE	
CLOSURE TEST RIG	
FOR SM-2 REACTOR	
LAYOUT NO.	AEL-519
AEL-119

LAYOUT			
ALCO PRODUCTS INC. SCHENECTADY, NEW YORK ATOMIC ENERGY DEPT.			
MODEL SM-2	STARTED 10/15	DATE 10-15	BY J. E. Boyd
REFERENCES AEL-519	DATE	APPROVED	
AUTHORITY	DATE	APPROVED	
TITLE-CLOSURE VESSEL TEST SET-UP			
LAYOUT NO. RE-31-1PE	AEL-525		

D. METALLURGICAL DEVELOPMENT PROGRAM

1. Progress (Task 4.0) - R. D. Robertson

Type 304 stainless steel has been selected as the reference SM-2 pressure vessel material.

Type 403 stainless steel has been selected as the reference stud and nut material, with A-286 as a backup material. Physical property and corrosion testing of these two materials is scheduled.

Silver-Cadmium-Indium has been selected as the alternate absorber material. It is planned that a fabrication development program will be initiated if further investigation supports selection of this material.

A series of test requests are being prepared for SM-2 fuel element testing in other reactors, for burnup of SM-1 Core I elements to SM-2 levels and for examination of the highest burnup SM-1 Core I element at end of core life.

Prices are being solicited on types 347 and 304L stainless steel made to restricted chemistries.

Work was suspended on September 9, 1959, on corrosion testing of irradiated boron stainless steel. The facility is being maintained in its present nearly-completed state pending directions to dispose of the equipment and specimens or to continue the program.

BMI progress and the irradiation testing program are summarized separately.

2. Preliminary Evaluation and Planning of Metallurgical Program (Task 4.1) - A. S. Wilder

Requests are being made to insert fuel elements with SM-2 concepts in the SM-1 and the MTR or ORR, for the continued use of SM-1 Core I fuel elements and absorbers in SM-1 Core II, for examination of the highest burnup SM-1 Core I element at end of core life, and for periodic examination of the core during Core II life. Information obtained in this manner will supplement information obtainable from other sources.

3. Type 347 Stainless Steel Investigation (Task 4.2) -
R. W. Kelleman, G. P. Pancer and J. L. Zegger

Based on the recommendations made in APAE Memo No. 214 regarding cobalt and tantalum restrictions in type 347 stainless steel, new inquiries were made with the various steel mills supplying this material. This was necessary since the chemical restrictions vary somewhat from those initially specified.

Dose rate calculations indicate that for the fuel element cladding, flow divider and skirt, type 347 with 0.025 maximum cobalt and 0.01 maximum tantalum will be required. However, if for reasons of economics or availability, type 347 with the above restrictions cannot be supplied in the size and shapes necessary for the flow dividers and skirt, then type 304L with 0.025 maximum cobalt can be used as a substitute. Other core components such as the reflector rings, upper and lower grid plates, tie rods, etc. will utilize type 304L stainless steel material with the cobalt restriction relaxed to 0.04 maximum, if there is a significant price differential between 0.04 and 0.025 maximum cobalt in material of this type.

It is expected that prices on both type 347 and type 304L with varied chemistries will be received during the next reporting period.

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

Work on corrosion testing of irradiated boron stainless steel samples procured from KAPL was discontinued as of September 9, 1959 due to suspension of this program. At the time work ceased, approximately \$7,000 had been spent on this project, and the corrosion testing facility was nearly completed. The facility will be maintained in its present state pending directions to continue work or to dispose of the equipment and specimens. Other possible uses of this equipment are being considered. One autoclave is currently being used for corrosion testing of Eu_2O_3 absorber specimens.

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

Due to some difficulties encountered in editing the report covering the effects of neutron irradiation on the properties of structural materials, the rough draft has not yet been submitted for final typing. However, it is expected that final editing will be completed within two weeks.

The reference material initially considered for the SM-2 reactor vessel was an austenitic stainless steel type 347. This material appeared necessary

because of the possibility of the pressure vessel wall attaining a temperature as high as 650°F. At this temperature, type 347 is preferable to type 304 because of the possibility of carbide precipitation in type 304 after long term exposure. However, recent thermal analyses indicate that the maximum temperature the reactor vessel wall will attain in service is 550°F. At this temperature, the possibility of carbide precipitation in type 304 is negligible. Consequently, the austenitic stainless steel reference material has been changed from type 347 to type 304.

The irradiation damage resistance and corrosion resistance of type 304 is similar to that of type 347, but the ease of fabrication is improved. Confidence in obtaining a properly constructed vessel is also increased as cracking problems anticipated in welding heavy sections of type 347 are eliminated by the use of type 304.

6. Nut and Bolt Materials for Stainless Steel Reactor Vessel
(Task 4.5) - R.W. Kelleman

Stress calculations have been made on various nut and bolt material considered for use in the SM-2. Although these calculations indicate A-286 to be a desirable material, it could not be recommended as reference, since it is not ASME code approved. Therefore, type 403 has been selected as the reference bolting material with A-286 as a back-up material.

It is planned to conduct testing on these two materials to check their physical properties and corrosion resistance. Bids are now being obtained for the material which will be required for the closure material development and testing program. Procurement of this material will be made upon approval from the contracting agency.

7. Alternate Absorber Material (Task 4.5) - A.S. Wilder

Europium oxide, the reference SM-2 absorbing material, has two basic disadvantages. High temperature water in direct contact with Eu_2O_3 , such as would occur in the event of a clad failure, may cause gross swelling of the absorbing material or transport of high activity europium through the primary system. Furthermore, europium oxide is quite expensive (~\$1400/kg). As a result, a literature survey was made in an effort to select a desirable alternate reference material. All materials were evaluated on the basis of:

1. Nuclear acceptability.
2. Metallurgical fabricability

3. Corrosion resistance
4. Resistance to irradiation damage
5. Effect on primary system activity
6. Economics
7. Availability

Based on these criteria, Silver-cadmium-indium appears to be the most acceptable for use in the SM-2 system.

From a nuclear standpoint, Ag-Cd-In is a fairly good absorbing material. While Eu_2O_3 in the same size slab is expected to have longer life, the Ag-Cd-In should have adequate life for a least one core loading. Each of the three elements has a nuclear shortcoming; however, in combination they provide effective coverage. Burnup of the absorber will have to be further investigated to determine actual slab thickness.

Ag-Cd-In is readily formed into any desired configuration. At SM-2 temperatures the creep strength is low; therefore the material should be structurally supported. Ag-Cd-In cannot be roll clad with stainless steel, however canning probably can be done with no undue difficulty.

Corrosion resistance may be very poor under some water conditions.

For these reasons canning with stainless steel or plating is desirable. Properly canned it is expected that both resistance to corrosion and creep will be adequate for SM-2. Study of canning technique, and such problems as allowable gap between canning and absorber, is expected to be investigated in the next reporting period under the expanded SM-2 program.

There is no indication that resistance to radiation damage is any problem. All three elements undergo an (n, γ) reaction upon neutron absorption; therefore, there is no gas evolution such as is found in absorbers containing boron alloys or dispersions. Inclusion of samples in the irradiation capsule program is being considered.

Absorber activity would be several times higher than europium oxide at shutdown, but would decay rapidly to less than Eu_2O_3 after 30 days.* If canning failure occurred, the activity would be transported throughout the primary system. Below 400°F the Ag-Cd-In tends to plate out on the system but above this would remain in solution. Cleanup of the system could be a serious problem. Further work on activity and transport of activity is planned.

Ag-Cd-In is both cheap (\$25-40/lb) and available. Consideration of hafnium was dropped when investigation revealed that it was not available.

* Anderson, Kermit, "Broad Aspects of Absorber Materials Selection for Reactor Control," paper presented at meeting of American Nuclear Society, New York, October 27-30, 1957.

E. WELDED FUEL ELEMENT DEVELOPMENT

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

The TIG process has been selected as the reference method for welding elements.

No crater cracking has occurred since changing from 0.030" to 0.040" side plates.

Since the last quarter, TIG welded fuel elements have been fabricated with spacers between fuel plates to minimize distortion during welding, resulting in improvement in channel spacing tolerances.

Elements have been given various thermal cycles to 650°F and almost no distortion, rippling, or channel spacing change has occurred.

An element has been welded using two depleted fuel plates with canted cores. Micro and macroscopic examination by BMI indicated that a minimum of 0.050" fuel plate dead edge would be satisfactory for welding.

A manufacturing procedure is being written for welded SM-2 fuel elements.

Static corrosion tests have been made and evaluation of results have been reported on two TIG welded dummy fuel elements. The TIG welded elements appeared to resist corrosion adequately.

Specimens of Eu_2O_3 dispersion both unclad and clad (with simulated defects) from BMI and a specimen from ORNL were static corrosion tested at 635°F in simulated reactor coolant.

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

Elements welded with the TIG process using spacers between fuel plates have produced elements with channel spacings within 0.005" average variation. No problems were encountered with spacer removal.

Since electrical conductivity is not a requirement of spacers with TIG welding as in resistance welding, new spacers have been ordered which will be chrome plated steel made to closer tolerances.

To date, eight elements have been TIG welded using 0.040" thick side plates. No crater cracks or other defects have appeared in 2,224 welds.

In order to evaluate fuel plate minimum dead edge, a partial (2 plate) fuel assembly was welded using depleted plates with offset cores obtained from Sylcor. The cores in these fuel plates were offset so that dead edge widths varied from 0.040" at one end to 0.100" at the other end. Reference TIG welding conditions were used in assembly. The partial fuel assembly was then sent to BMI for metallurgical examination and evaluation.

No detrimental effects were found on the fuel plates due to the welding operation. No core distortion, plate warpage or non-uniform spacing outside of normal tolerances could be detected. There was no evidence of microporosity, void areas, slag inclusions in the welds, or puddling of UO_2 , even in welds made in sections containing the 0.040" dead edge.

A minimum of 0.050" dead edge is recommended, however, because of variation in edge effects and the difficulty in detecting the extreme edge of the core (the last UO_2 particle) by radiography or other inspection techniques. A minimum dead edge width of 0.050" has therefore been selected for reference SM-2 fuel plates for assembly by welding. An increase in meat width of approximately 7% will be gained by this reduction of dead edge. This increase in heat transfer area will have an important effect on core volume flow.

From this same element, one depleted cold rolled and one solid plate were removed for thermal stress tests under Task 6.

The following elements have been thermal cycled at various rates. On an average there has not been more than 0.001" change in channel spacings and no measurable rippling has been detected.

<u>Element</u>	<u>Description</u>	<u>Cycle</u>	<u>Change</u>
Extra	0.040" solid plates, TIG welded	Injected into furnace @ 650°F - held 1/2 hr. - air cooled	0.0005" avg. change in channel spacings No rippling
		3 cycles to 600°F @ 100°F/hr	0.0005" avg. change in channel spacing No rippling
#7	0.040" depleted annealed plates, TIG welded	3 cycles to 650°F @ 125°F/hr	Negligible channel change. No rippling see Fig. 1.

Rippling measurements taken to date on welded fuel elements are summarized in Figure 5-1. If plate end measurements are disregarded, the maximum variation is about 0.006" and the average variation about 0.002". Plate end measurements are believed to reflect the lack of stiffening combs and the effect of handling. If plate end measurements are included, the maximum variation is about 0.010".

3. Corrosion Testing (Task 5.2) - C.R. Bergen

Two unclad and two clad Eu_2O_3 dispersion specimens (with simulated defects) were tested at 635°F. These first BMI preparations showed swelling and distortion in as few as three hours, indicating a lack of isolation of the individual granules of Eu_2O_3 . Full details of this study can be found in AP Note 184.*

A defected clad specimen was also obtained from ORNL. This specimen had shown no damage during 72 hours at ORNL in 575°F water although it was drilled through to simulate cladding defects. Exposure for 8 hours at 635°F at Alco produced no distortion and only a slight weight loss attributed to leaching of Eu_2O_3 from the exposed dispersion surface.

These preliminary tests indicate that corrosion resistance of Eu_2O_3 is a function of crystalline form.

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

a. Welded Fuel Elements

A concentrated effort will be made on the writing of the manufacturing procedures for the TIG welding of fuel elements.

Four elements will be TIG welded, two using cold rolled, 0.040" depleted plates, and two using annealed, 0.040" depleted plates, all having a nominal 0.050" dead edge.

These four elements will be checked microscopically and corrosion tested.

b. Corrosion Testing (C.R. Bergen)

A static autoclave test is underway to determine if the possibility exists for extension of crater cracks by stress corrosion. Results of this test should be available in about two months.

* Bergen, C.R., "Effect of Exposure of Defective Clad Eu_2O_3 Absorbers to Simulated Reactor Coolant Conditions, A Preliminary Study," AP Note 184, Dec. 12, 1959.

Dynamic corrosion testing of TIG welded fuel elements will begin in February 1960. This test will include static testing of half elements as a control or cross check. Cold rolled depleted elements in full length will also be tested during this period. Results of the dynamic phase will be available in June 1960.

Corrosion studies will be continued on BMI prepared clad Eu_2O_3 dispersions as these become available. A long term corrosion test will be run on the available ORNL specimen. Analytical techniques will be developed to pick up low levels of Eu dispersed in autoclave water, as an aid to picking up defects in clad absorber elements.

12/16/59

**SM-2 CORE MATERIALS
DEVELOPMENT PROGRAM**

November 23, 1959

**BATTELLE MEMORIAL INSTITUTE
505 King Avenue
Columbus 1, Ohio**

622 079

SM-2 CORE MATERIALS
DEVELOPMENT PROGRAM

BATTELLE MEMORIAL INSTITUTE

November 23, 1959

I. Program Objectives

The general objective of the program is to develop fuel, burnable-poison, absorber, and suppressor materials for the SM-2 core. This report reviews work performed from September 4 to November 23, 1959. Studies concerned with the compatibility of materials, determination of physical and mechanical properties, development of fabrication processes, and evaluation of fuel and burnable-poison materials under irradiation are being conducted. Based on these studies, material specifications for the fuel, burnable poison, absorber, and suppressor will be established by June 30, 1960.

A more explicit list of the program objectives and current aims of the research and development studies is presented below.

- (1) Study compatibility of boron and boron compounds in stainless steel. This includes the determination of factors that influence the boron-stainless steel reaction and their effect on this reaction and on the subsequent loss of boron. Factors under consideration include (a) composition of the stainless steel matrix, (b) impurities in the matrix and boron compound, (c) sintering atmosphere, (d) fabrication schedule, and (e) composition of boron compounds. It is anticipated that from these studies an understanding of the mechanism that is involved in the deboronization of stainless steel will evolve. The coating of the boron compounds to eliminate or suppress reaction of the compound with the stainless steel matrix is also under study.
- (2) Evaluation of base materials. This includes the determination of properties of the UO_2 , Eu_2O_3 , and ZrB_2 , and correlation of the properties with the fabrication process. Based on these studies, specifications will be established for each material.
- (3) Determination of the effects of fabrication variables on the properties of the reference fuel plate.
- (4) Development of techniques for the fabrication of full-size reference fuel plates. This includes the study of physical and mechanical properties, determination of boron retention, and establishment of dimensional tolerances. Results of these studies will be utilized for the preparation of fabrication specifications.
- (5) Development of techniques for fabrication of stainless- Eu_2O_3 fuel plates. Results of studies performed at Oak Ridge, Alco, and Battelle are being correlated. Studies at Battelle include evaluation

of pretreatment of Eu_2O_3 and development of reference fabrication process.

- (6) Drafting of preliminary material and fabrication specifications. This will delineate areas of weakness and serve as a guide for future studies.
- (7) Preparation of irradiation test specimens. Fabrication has been completed; analysis for boron is in process on several specimens.
- (8) Evaluation of irradiated mock-up capsules to obtain a better estimate of flux perturbation. Results will be utilized for predicting burnup obtained for specimens irradiated in the MTR and ETR capsules.
- (9) Design of irradiation capsules. The design of both MTR and ETR capsules has been completed; however, failure of heaters in the ETR capsule necessitates modification of remaining six ETR capsules.
- (10) Fabrication of irradiation capsules. Fabrication is to be completed on six ETR capsules. Laboratory tests and evaluation of operational data are being made to determine the cause of heater failure in the first capsule. Results from tests and evaluations will be used as the basis for modification of the initial design.

II. Current Status of Work

The current schedule for the program is presented in Figure 1.

An extension of time for the irradiation of the MTR capsules from February to March is being considered. The time extension would make up the time lost during the recent period of activity in the MTR when the capsules were discharged temporarily from the reactor core. It is contingent on the time available to ship the capsules to Battelle and complete the necessary postirradiation evaluations by June, 1960.

During this report period, the auxiliary heaters incorporated in the first ETR capsule failed. Laboratory tests are being conducted and an evaluation of the operation and design of this capsule is being made to determine the cause of failure. This has delayed the time for fabrication of the remaining six capsules and will also delay the time these capsules are inserted into the ETR.

III. Materials Development

Fabrication techniques are being evaluated for use in developing processes for preparing fuel plates, suppressor components, and absorber plates for the SM-2 reactor. Reference materials have been selected, and specifications will be established on the basis of the fabrication studies.

FIGURE 1. CURRENT SCHEDULE OF EXPERIMENTAL PROGRAM

Irradiation specimens have been fabricated for all tests presently scheduled, and major emphasis is now being placed upon the production of full-size fuel elements for use in critical-assembly tests and welding studies.

A. Fuel-Plate Fabrication

Twenty full-size reference fuel plates containing cores of 26 w/o fully enriched UO_2 and 1.2 w/o ZrB_2 dispersed in Type 347 prealloyed stainless steel (0.030 in. thick) and clad with Type 347 stainless steel (0.005 in. thick) are being prepared for the critical-assembly tests. Forty plates which are similar but contain depleted UO_2 are being prepared for the welding studies.

As a check on boron content, small-scale specimens are being sintered and rolled simultaneously with the full-size cores, and these specimens are being analyzed for boron content. A small compact sintered (2 hr at 2150 F) with the first series of fully enriched cores contained 0.225 w/o boron by analysis, which is approximately 3 per cent greater than the calculated amount added to the mix. This value is within the range of accuracy for chemical analysis (± 5 per cent). A small roll-clad specimen prepared at the same time is being analyzed, but the results are not yet available.

All plates fabricated to date are well within the prescribed specification. Dimensional data on the first seven plates produced for the critical assembly are listed in Table 1. Actual dimensions listed in the tentative specifications are as follows: core length, $22 - 1 + 0.5$ in.; core width $2.625 + 0.089$ in., including camber; plate thickness, 0.040 ± 0.001 in.; cladding thickness, 0.005 ± 0.001 in. No data on the cladding thickness will be available on these plates until all fabrication is complete. However, examination of depleted plates made by the same process indicates that this dimension will be maintained. The core length has been deliberately maintained at >22.0 in. because this length is desired for the studies to be conducted at Alco.

TABLE 1. DIMENSIONAL DATA FOR SEVEN FULLY ENRICHED SM-2 REFERENCE FUEL ELEMENTS

Plate	Plate Thickness, in.	Core Length, in.	Core Width, in.	Camber, in.
EAR-2	0.0405	22.5	2.65	0.052
EAR-3	0.0410	22.5	2.64	0.032
EAR-4	0.0405	22.3	2.64	0.055
EAR-5	0.0405	22.5	2.64	0.016
EAR-6	0.0400	22.5	2.64	0.035
EAR-7	0.0405	22.5	2.64	0.032
EAR-8	0.0400	22.3	2.65	0.020

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B. Boron-Retention Studies

A high-purity fused grade of ZrB_2 was utilized for the preparation of the enriched and depleted fuel plates. The ZrB_2 was procured from the U. S. Borax and Chemical Corporation at a cost of \$10.50 per lb. The as-received powder was minus 325 mesh and contained 19.9 w/o boron.

The desired ZrB_2 size for the reference SM-2 plates is minus 100 plus 200 mesh. The minus 325-mesh powder was consolidated by the following process:

- (1) A batch of 20 g was cold pressed in a 1 by 1-in. die at 15 tsi.
- (2) The pressed compacts were arc melted in a gettered inert-gas atmosphere. A single fusion melt was used to minimize boron loss and pickup of gaseous impurities.
- (3) The fused compact was lightly pickled with acid to remove any oxide film.
- (4) The compact was crushed and screened to minus 100 plus 200 mesh.

After melting and crushing, analysis of three individual batches indicated 18.8, 19.3, and 19.2 w/o boron. The behavior of this material under fabricating conditions may be seen by referring to the photomicrographs in Figures 2 and 3. As shown in Figure 2, even when it is etched to show boron-containing phases, there is no evidence of any reaction either in the ZrB_2 or the matrix in the as-sintered compacts. Chemical analysis of a compact in this series indicated a boron content of 0.176 w/o compared to the 0.179 w/o boron originally added to the mix before sintering. This value (a decrease of 1.7 per cent total boron) is well within the range of accuracy for these analyses.

Although the use of vacuum sintering has made it possible to prevent any measurable loss of boron during the sintering process, there is some indication that a slight reaction occurs during the rolling operation. As shown in Figure 3, examination of an unetched roll-clad specimen shows that a thin oxide layer forms on the ZrB_2 particle during the rolling operation; but there is no indication of a reaction. However, in the etched specimen in Figure 3, which also shows numerous carbide particles and other inclusions in the steel, a pinpoint precipitate believed to contain boron may be detected around the ZrB_2 particles. The precipitate, which is lightly outlined in the photomicrograph, extends approximately 1 to 2 mils from the ZrB_2 particle. Based on a chemical analysis, the boron loss was estimated as 5 per cent of the total boron. An exact value could not be obtained since a portion of the specimen was used for metallography.

Losses of up to 12 per cent of the boron have been reported on small-scale specimens roll clad by standard procedures. During the roll-cladding operation, the ZrB_2 particles are exposed to a greater surface of stainless steel and, consequently, to the impurities within it. It is believed that the major difficulty may be due to reaction with oxygen, since studies on other impurities have shown no definite trend toward a reaction. There is, however, no indication of a boron diffusion gradient through the cladding.

A photomicrograph of the core-cladding interface is shown in Figure 4. It should be noted that grain growth across the interface is complete; there is no preferential distribution of impurities or secondary phases at the interface, and there is no indication

250X

As Polished

N61772

500X

Electrolytic NaCN Etch

N61771

FIGURE 2. AS-SINTERED FUEL-ELEMENT CORES CONTAINING 26 w/o UO_2 -1.0 ZrB_2 IN A PREALLOYED TYPE 347 MATRIX AFTER SINTERING 2 HR AT 2150 F IN VACUO

FIGURE 3. CORE STRUCTURE OF FUEL ELEMENT CONTAINING HIGH-PURITY ARC-CAST ZrB_2

This specimen, containing 1.0 w/o ZrB_2 -26 w/o UO_2 in a Type 347 stainless powder matrix, was sintered and roll clad at 2150 F.

100X

N63311

500X

N63312

FIGURE 4. CORE-CLADDING INTERFACE ON FUEL ELEMENT FABRICATED BY REFERENCE PROCEDURE

622 087

of a boron-containing phase in the cladding. In general, the only way of distinguishing between the core and cladding is by the grain size, which is always less in the core.

A study has been made of the retention of boron in stainless-boron alloys during high-temperature heat treatment in varying media. A stainless-0.461 w/o boron alloy in the form of 0.010-in. -thick foil was utilized in these studies.

Boron losses are rapid and severe when the foil is annealed in hydrogen at 2000 F. As shown in Figure 5, the disappearance of secondary boron-containing phases is apparent. However, it is difficult to observe any boron concentration gradient in the alloy when the boron content is too low to cause precipitation of a secondary phase. It is thus possible for small amounts of boron to diffuse through the cladding without being observed metallographically. Boron analyses for the foils are also listed in Figure 5. It will be observed that, upon acid pickling to remove the surface layer, the boron content of the hydrogen-annealed specimen decreased while the boron content of the vacuum-annealed specimen increased. Based on work by various researchers on diffusivity of boron in commercial steels, it can be hypothesized that when sufficient oxygen is present, as in the hydrogen atmosphere, not only does the boron diffuse out but oxygen diffuses in and reacts with boron below the surface. This boron oxide may diffuse slowly to the surface and be lost, but it does not greatly interfere with the normal boron diffusion. Thus a higher concentration of boron will be present just below the surface than at a point closer to the center of the foil. In the vacuum-annealed specimen one would expect a small amount of oxygen to be available to react at the surface, but not a sufficient amount to diffuse into the metal. Therefore, a depleted boron surface would be expected. It should also be noted that, because of varying oxygen contents, the behavior of the foil in hydrogen is erratic and in some cases almost complete removal of the boron has been observed.

Although losses of up to 12 per cent of the contained boron were observed in rolled fuel plates, the average loss has been about 5 per cent. It is possible to achieve a specified ± 5 per cent boron content by use of a 10 per cent excess of boron in the preparation of the fuel compacts. However, the objective of these studies is to develop a process which will not be dependent on the use of an excess boron loading. As these studies progress and a better understanding of the stainless-boron reaction and stainless deboronization process is obtained, there is every indication that this objective will be achieved.

C. Study of Fuel-Plate Fabricational Variables

Small-scale compacts are being used to evaluate fabrication variables such as rolling temperatures, particle sizes of UO_2 , ZrB_2 , and stainless powders, and reduction ratios and rolling schedules. In addition to metallographic examinations and chemical analyses, transverse tensile tests are being used to evaluate core and core-cladding bond strengths.

In order to further study the retention of boron during rolling, a series of specimens is being prepared which will be used to determine the effect of time and temperature of the various operations on the boron loss. In addition, an attempt is being made to coat ZrB_2 particles with chromium, niobium, and/or tungsten by a fluidized-bed process. These particles will be used to determine whether a protective coating will be beneficial in preventing secondary reactions with impurities in the stainless matrix.

Hydrogen Annealed

The specimen contained 0.252 w/o boron as annealed and 0.182 w/o boron after pickling to remove surface layer.

Vacuum Annealed

The specimen contained 0.395 w/o boron as annealed and 0.432 w/o boron after pickling.

Hydrogen Annealed

Vacuum Annealed

FIGURE 5. COMPARISON OF BORON-CONTAINING ALLOYS (0.461 w/o BORON) AFTER 2-HR ANNEALS AT 2200 F

The top photomicrograph shows the center region of the specimens, while the bottom photomicrograph shows regions near one end of the specimens. Notice the badly depleted area in the hydrogen-annealed specimen.

Results of recent transverse tensile tests are shown in Table 2. All plates containing spherical UO_2 possessed higher strengths than those primarily reported for similar plates containing high-fired UO_2 . Increases in strength of as much as 13,000 psi were obtained by using spherical UO_2 . Although not shown in the table, another interesting observation was made which showed that a full-sized fuel element containing high-fired UO_2 had a room-temperature transverse tensile strength of 25,100 psi, while a similar small-scale specimen had a transverse tensile strength of 17,000 psi. Additional specimens will be run to determine whether such behavior is consistent.

TABLE 2. RESULTS OF TENSILE TESTS IN THE THICKNESS DIRECTION OF FUEL PLATES CONTAINING SPHERICAL UO_2

UO_2 , w/o	ZrB ₂ , w/o	Reduction on First Pass, per cent	Core Thickness, in.	Core Width, in.	Total Reduction Ratio	Test Temperature, F	Ultimate Tensile Strength, psi
26	1.45	12	0.090	1	1.1:1	Room	22,800
26	1.45	12	0.090	1	1.1:1	600	20,300(a)
26	1.45	19	0.092	1	1.2:1	Room	23,700
26	1.45	36	0.097	1	1.5:1	Room	29,500(a)
26	1.45	44	0.100	1	1.8:1	Room	29,700(a)

(a) Average of two tests.

Small-scale specimens are also being used to determine the effect of fabrication temperature and particle size of powders. Metallographic examination of a series of rolled compacts pressed and sintered at temperatures of 2000 to 2200 F indicates that all core structures containing spherical UO_2 are superior to those containing high-fired and other types of UO_2 fabricated at the 2200 F temperature. However, there is a definite improvement even in the spherical UO_2 dispersions as the temperature is increased to 2200 F. At 2000 F, the UO_2 is fractured even though it resists stringering, but at 2200 F, there is relatively little fracturing and virtually no stringering. Some improvement is noted in structures rolled at 2000 F when higher sintering temperatures are used.

A series of cores containing 26 w/o UO_2 dispersed in matrices of 100 w/o minus 44- μ -diameter stainless powder, 100 w/o 105 to 150- μ -diameter stainless powder, and 20 w/o 105 to 150- μ -diameter-80 w/o minus 44- μ -diameter stainless powder were fabricated by sintering and roll cladding at 2200 F. Metallographic investigation of the roll-clad cores indicated that an improved structure was obtained with the minus 44- μ -diameter stainless powder. The use of 100 w/o 105 to 150- μ -diameter matrix powder resulted in considerable fracturing and stringering of the UO_2 , while the 20 w/o 105 to 150- μ -diameter-80 w/o minus 44- μ -diameter powder exhibited slightly less fracturing and stringering of the UO_2 shot.

D. Development of UO_2 Fuels

Several grades of spherical UO_2 manufactured by Mallinckrodt and Numec have been received and evaluated by petrographic analyses. In addition, the reference fuel, Mallinckrodt fully enriched UO_2 , has been analyzed for oxygen content and evaluated on the basis of X-ray diffraction. The results of the petrographic analyses are shown in

Table 3. The Numec material has not been evaluated from a fabrication standpoint, but from the petrographic analysis it appears to compare favorably with the Mallinckrodt material. It should be noted that the reference material was made in Mallinckrodt's new equipment for producing large quantities and that it appears to have the characteristics desirable for dispersion fuels. Impurities as determined by spectrographic analyses are as follows:

Element	Amount, ppm	Element	Amount, ppm
Aluminum	10	Lead	<5
Arsenic	20	Magnesium	5
Barium	<5	Manganese	2
Beryllium	0.5	Molybdenum	<10
Boron	<0.5	Nickel	<5
Cadmium	<0.4	Phosphorus	<40
Calcium	5	Samarium	<5
Chromium	10	Silicon	40
Cobalt	<2	Titanium	<20
Copper	7	Tungsten	<60
Gadolinium	<5	Vanadium	<5
Iron	55	Zirconium	<20

Wet chemical analysis gave an oxygen-to-uranium ratio of 2.01 and a carbon analysis of 0.003 w/o. A lattice parameter of 5.470 Å, which is very nearly correct for stoichiometric UO_2 , was obtained from X-ray diffraction data. No secondary phases were detected. It will be noted that the petrographic analyses indicate a secondary phase. In general, this phase is a black opaque vitreous material found in the grain boundaries. Its composition is unknown, but it probably is slightly enriched in uranium. It should also be pointed out that this phase, which can also be considered as a thick grain boundary, is responsible for the high strength of the particles. The reference material also has very small grains. According to the present state of knowledge this characteristic is favorable from the standpoint of fabrication.

TABLE 3. PETROGRAPHIC ANALYSES OF SPHERICAL UO_2 POWDERS

Material	Phase Identity, volume per cent		Grain Size, μ		Pore Space, volume per cent	Equivalent Water Absorption, per cent	Rating of Crushing Strength ^(b)
	Isotropic UO_2	Phase B ^(a)	Average Diameter	Maximum Diameter			
Mallinckrodt ^(c) enriched shot, A-491	90.4	9.6	7.2	28.5	3.5	0.9	1
Mallinckrodt natural shot, A-473	86.3	13.7	1.6	2.9	0.15	0.04	1
Mallinckrodt depleted shot, A-488	92.0	8.0	13.2	70.5	3.0	0.8	2
Numec natural shot	91.4	8.6	18.3	85.8	2.62	0.66	3
Numec special spherical shot	91.6	8.4	12.4	42.2	2.7	0.7	1

(a) Black opaque vitreous phase usually occurring in grain boundaries.

(b) Relative values with 1 being the highest strength.

(c) Reference material used for critical studies.

The values for crushing strength are qualitative and were obtained in the following manner. A small sample of the as-received shot was placed in an agate mortar and ground with an agate pestle for 5 min. A representative sample of the pulverized powder was then inspected under the petrographic microscope. The remaining part of the sample was ground for an additional 5 min and inspected. The rate of physical breakdown at 5 and 10 min was noted and compared for the various samples.

The porosity was measured by direct count of internal voids in areas of more than 100 particles of each material. The equivalent percentage of water of absorption is calculated from the volume of pore space on the basis of experimental results with other ceramics. This value is simply an estimate of the open porosity.

E. Fabrication of Irradiation Specimens

All irradiation specimens for the SM-2 irradiation program as revised June 25, 1959, have been fabricated, leak tested, and radiographed. The boron analyses and metallographic examinations of representative samples of each specimen type, fabricated in the last group, are now in process. A tentative assignment of specimens to the ETR capsules is listed in Table 4.

TABLE 4. TENTATIVE GROUPING OF SPECIMENS FOR THE ETR CAPSULES

Specimen Types to be Given 38 Per Cent Burnup				Specimen Types to be Given 77 Per Cent Burnup		
3(a)	1	1	3	5(b)	1	1
4	4	5	5	6	4	4
6	13	7	6	9 (11)	9 (11)	5
7	14	13	9 (11)	13	13	6
9 (11)	8	8	12	16	16	17
12	--	--	14	18	17	18

(a) Capsule BMI-32-4.

(b) Capsule BMI-32-5 specimens are awaiting encapsulation.

As each group of specimens is assigned to an irradiation capsule, each specimen is notched on the top edges to facilitate postirradiation identification. Both sides of each specimen are then photographed at magnifications of 1X and 4X.

F. Development of Absorber Materials

Dispersions containing 33 w/o Eu_2O_3 in an 18-11 elemental stainless steel powder matrix were made and corrosion tested at Alco. These specimens measured 1 by 4 in. and contained a 0.090-in. -thick core clad with 0.010-in. Type 347 stainless steel. A silicon-free stainless steel foil approximately 0.005 in. thick before rolling was used as a barrier between the core and cladding. A representative microstructure of these specimens is presented in Figure 6. The specimens were defected by holes and machined grooves to simulate cladding failure.

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100X

RM12874

FIGURE 6. LONGITUDINAL SECTION OF Eu_2O_3 -ELEMENTAL STAINLESS DISPERSION

The specimens exhibited an approximate 30 per cent swelling after a 24-hr corrosion test in high-temperature pressurized water. However, there were no ruptures detected in either the matrix or cladding-to-matrix interface. A hydration of the Eu_2O_3 particles was observed in areas in the vicinity of the defect which were directly exposed to the water.

Recent tests conducted at ORNL revealed only slight hydration in the defected areas and no detectable swelling. As a result of this difference in behavior, additional corrosion studies are planned. Dispersions are presently being prepared containing Eu_2O_3 prepared by three different techniques. These dispersions will be roll clad and corrosion tested. It is anticipated that through these studies, the optimum product will be selected and the cause of swelling of the initial corrosion-test specimens will be determined.

One series of specimens will contain Eu_2O_3 powder prepared in the same manner as used for the initial corrosion specimens. Powder of minus 325-mesh size is mixed with a Ceremul "C" binder, dried at 150 F for 1/2 hr, pressed at 6 tsi, and crushed to minus 30 plus 80 mesh. These particles are then sintered in a platinum boat at 2750 F for 3 hr. The sintered material is crushed and screened to the desired size.

Another technique of preparing Eu_2O_3 powder is the technique employed by ORNL. The powder is pressed with no binder at 4 tsi and the pellets are fired at 3090 F for 3 hr in a hydrogen atmosphere. The sintered pellets are then crushed and screened to size.

The third type of Eu_2O_3 to be used in the comparison was obtained in the desired particle size from a commercial source. This powder was fired in air at 3270 F without pelletizing and with no binder. The particles produced by this process have a smooth fused surface and a porosity of <0.1 per cent. The thoria content is <0.01 per cent, as reported by the supplier.

This powder is presently being evaluated by X-ray diffraction, chemical, and petrographic analyses. As an additional improvement on corrosion resistance, it is thought that some benefit may be derived by coating the Eu_2O_3 particles by the vapor-deposition process. It is therefore planned to make an additional evaluation study. A protective coating will be vapor deposited on each particle of Eu_2O_3 before blending with the stainless steel matrix. This blend will then be compacted and roll clad for comparison with dispersions containing uncoated particles. The fabrication techniques will also be varied during the evaluation studies to determine the optimum process for each material.

IV. Status of Irradiation Program

Four capsules containing clad fuel specimens of interest to the SM-2 program are being irradiated at the MTR and the ETR.

The design of these capsules has been described in previous reports of this series. In each case, the capsule heat-transfer characteristics were selected to achieve a specimen-surface temperature of approximately 600 F in an effective specimen thermal-neutron flux of approximately 1.5×10^{14} nv.

Of the four capsules, three are noninstrumented and are being irradiated in core positions at the MTR. The fourth capsule is equipped with thermocouples and heaters and is being irradiated in a beryllium-reflector position at the ETR.

A. Irradiation History of MTR Capsules

The status of the noninstrumented MTR capsules is as follows:

- (1) BMI-32-1 (contains seven specimens, including one double-length specimen with Eu_2O_3 suppressor) was inserted for Cycle 126 (August 15) in Core Position L-53, wherein the estimated unperturbed thermal-neutron flux in the specimen zone ranges from 3.4 to 4.2×10^{14} nv.
- (2) BMI 32-2 (loaded similarly to BMI-32-1) was inserted for Cycle 127 (September 6) in Core Position L-56, wherein the quoted unperturbed flux ranges from 3.1 to 4.9×10^{14} .
- (3) BMI-32-3 (loaded with eight standard-size specimens) was inserted for Cycle 128 (September 30) in Core Position L-58, wherein the estimated unperturbed flux ranges from 2.7 to 4.1×10^{14} nv.

During the first part of MTR Cycle 129, the activity of the water and the stack effluent was higher than normal. Also, unusually high concentrations of nickel were found. Since the three 32-series capsules are constructed with nickel outer shells, they were suspected of contributing to the situation. Hence, at a midcycle shutdown during Cycle 129, they, along with other capsules, were removed from the reactor. During the second half of Cycle 129, the activity persisted, indicating that the capsules discharged were, at least, not solely at fault.

After removal from the reactor, the 32-series capsules were examined visually both in the MTR canal and in the NRTS Hot Cell. The only unusual finding was that BMI-32-3 had a small dent in the weld joining the shell to the bottom header. This dent is the type that might be expected if the capsule had been dropped obliquely. While there was no evidence that the dent had opened the seam and produced a NaK leak, reactor personnel did not consent to reinserting the capsule for Cycle 130 (November 6), feeling that additional checks were warranted. However, BMI-32-1 and BMI-32-2 were reinserted for Cycle 130.

At the time of this writing, the scheduled date of reinsertion of BMI-32-3 is Cycle 131 (November 27). Adherence to this date is, of course, dependent on not finding evidence that the capsule is leaking. At present there is no reason to suspect that the schedule will not be met.

Figure 7 shows the nominal burnup schedule as presently estimated for the specimens in these three capsules. The burnup levels which appear on the plot are based on reactor-quoted estimates of unperturbed flux and an empirically derived capsule and specimen flux-perturbation factor of 0.34. Nuclear mock-up perturbation data will soon be available and will permit a more refined estimate of rate of burnup.

B. Irradiation History of ETR Capsule

The instrumented capsule, BMI-32-4, was inserted in ETR Position 0-6 for Cycle 20 (September 14). The estimated peak unperturbed thermal-neutron flux in this position is 3.7×10^{14} nv. As described in previous reports, this capsule contains six standard-size specimens. Four of these are located in the top compartment of the capsule, which is furnished with 12 kw of auxiliary electrical-heat capacity to compensate for the reduction of specimen fission-heat generation as burnup progresses. Two specimens are located in a separate unheated compartment.

Capsule BMI-32-4 was shipped from Battelle to NRTS about mid-August. Before shipment, its heater and thermocouple system (a total of six 2-kw heaters and four thermocouples) was thoroughly checked and, at time of shipment, all units had proper continuity and exhibited excellent dielectric characteristics. The capsule lead tube was pressurized with helium for shipment to preserve these characteristics since experience had demonstrated that they were rapidly altered with absorption of moisture from air by the MgO used as insulation in the sheathed heaters and thermocouples.

Despite these precautions, one heater was found to be open just after arrival at the MTR. The reason for the failure could not be determined. No further losses of instrumentation occurred during the lead-tube bending step which was required to fit the capsule assembly into the irradiation position. However, shortly after the irradiation commenced, two of the four thermocouples failed. Although it may not be significant, it is of interest to note that both failed thermocouples are 0.062 in. in OD, the remaining two being 0.040 in. in OD.

After full reactor power was achieved, the thermocouple readings were:

FIGURE 7. CURRENT ESTIMATES OF BURNUP IN MTR CAPSULES

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Thermocouple 1 380 F

Thermocouple 2* 400 F

These levels correspond to specimen surface temperature of approximately 650 F. After approximately 1 kw of auxiliary electrical power was employed, the readings were:

Thermocouple 1 415 F

Thermocouple 2 430 F

The capsule temperature history to the end of October, indicated by these two thermocouples, is presented in Figure 8. The abnormal excursion of Thermocouple 1 during the first few days after the start of irradiation cannot be satisfactorily explained. In general, however, the thermal performance of the capsule has been very close to design conditions.

During mid-October, the five auxiliary heater units which were operative at the start of irradiation failed almost simultaneously and without warning. Resistance checks show in essence that one leg of each heater is shorted to the capsule body-lead tube assembly, which is grounded. It is not possible to pinpoint the exact reason for this failure, but it is conjectured that the dielectric characteristic of the resistance element-to-sheath potting material (a commercially available ceramic) used just above the capsule body has deteriorated. One constituent of this potting material is a silicate, a type of material that is subject to change of dielectric properties under irradiation.

Laboratory tests are being conducted at Battelle to investigate the possibility of spurious thermal breakdown of the suspected potting material. Various other elements in the sheathed heater-to-leadout wire connector system are also being re-examined to bolster potential weak links. At the present time, a heater element-to-leadout wire connector system of different design is being considered for use in the remaining ETR capsules. Actually, the development of this connector design paralleled that of the system used in the initial capsule. It has been used with good success in recent MTR capsule irradiations, although the number of individual heater leads in these capsules is less than is involved in the SM-2 capsules.

C. Nuclear Mock-Up Study

As indicated in the report dated September 3, 1959, it was desirable to conduct nuclear mock-up irradiations at the MTR and the ETR to check the Brad Lewis-derived perturbation factor employed in the capsule-design calculations. Late in August three mock-up capsules, using reject fueled specimens (based on 26 w/o UO₂) were shipped to NRTS; the capsules were irradiated in September and opened at the Battelle Hot-Cell Facility late in October. Final dosimeter data (by direct wire counting) are not yet available.

*The tip end of Thermocouple 1 is located opposite the face of the topmost specimen, being laid in a groove machined in the copper heat-dissipation block and separated from the specimen by 1/32 in. of NaK (to couple sheath). The tip end of Thermocouple 2 is similarly located opposite the face of the second specimen in the string. With the presently estimated fission-heat-generation rate, the estimated specimen surface-to-thermocouple temperature gradient is approximately 250 F.

FIGURE 8. MONITORED TEMPERATURES FOR BMI-32-4

The exposures of these capsules were conducted for approximately 15 min with the reactor at 1/100 full power. Exposure by position is as follows:

BMI-32-403 MTR Position L-58 (core)
BMI-32-404 ETR Position 0-6 (beryllium reflector)
BMI-32-405 MTR Position A-39-SW (aluminum reflector)

The latter experiment was conducted in anticipation of possible future interest in reflector positions at the MTR where an unperturbed flux as high as 2.5×10^{14} nv can be obtained.

V. Irradiation of Full-Scale Fuel Elements

The MTR is being considered for the irradiation testing of a modified SM-2 fuel element. The MTR staff has tentatively approved the irradiation of a test element in a core position assuming that the SM-2 element can be modified to obtain reasonable nuclear and hydraulic matching. It will not be possible to achieve a high burnup in the core position as the element must be discharged after the power generation is significantly reduced below that of a standard element.

After discharge from the core position, the irradiation of the element may be continued in one of the reactor lattice positions. However, there may be a delay to obtain adequate space as most suitable lattice positions are occupied.

DEL:pa

F. HYDRAULIC ANALYSIS AND TEST*

1. Progress (Task 6.0) - W.M.S. Richards, R.E. Williams**

Primary loop and control rod pressure losses have been estimated over a range of temperatures from an emergency startup of 40°F to operating temperature. No reduction in control rod static loadings determined for hot flow can be expected from the reduced system flow rates at low temperature.

Static deflection measurements were taken on a fuel element made up of cold rolled depleted plates. Test results indicate the deflection of these plates are about 18% lower than the annealed depleted plates.

Thermal deflection measurements were made on a brazed plate and a welded plate specimen. Both were taken from assembled elements. The brazed plate had initial ripples of 0.010". The amplitude of these doubled when a 110° temperature difference was imposed between plate centerline and plate edge. The welded plate after being cut out of the element had initial ripples of 0.015" and the amplitude approximately doubled under the estimated design temperature difference of 90°.

Plate vibrations were measured in a solid plate resistance welded element and a TIG depleted plate element. No significant vibrations were found up to 164% of the rated flow of 384 gpm.

Collapse measurements were taken on the above elements. With no leading edge comb, plate collapse was found in both elements in the range between 140% and 164% of rated flow. When leading edge comb was installed, no collapse was found.

Three tests were run to evaluate the latest core support plate configurations.

* Hydraulic tests are performed in the General Engineering Laboratory.

** This section of the report was written prior to selection of the latest reference flow presented in Task 2.

Inlet Plate Configuration	Exit Plate Configuration	Max. Range of Element Flow Distribution	Pressure Drop ft. H ₂ O	Remarks
AEL 393	AEL 393	+ 3% - 5.5%	19.7	Low Lattice Flow
AEL 523	AEL 393	+ 9% - 6.5%	17.9	1st Pass Design
AEL 523	AEL 526	+ 7% - 5%	41.4	2nd Pass Design
Target Values		+ 6 - 6	37 cold	2nd Pass Design

Top and bottom plate designs have been developed which provide adequate flow in the lattice and outside fuel-channel passages. The use of plate design AEL 523 for both the top and bottom core plates is definitely indicated from test results. This plate design will be especially effective with the new, uniform length, fuel plate element design (R9-13-1017) which does not require as much preferential flow treatment of the outside passages.

Flow distribution tests were made on the control rod. As presently designed, the outer channels are about 20% low in flow. Tests indicate that the proximity of the leading edge of the absorber is restricting the flow in these channels.

Detail design is proceeding on the full scale rig. Estimated completion date for the testing is about the end of September 1960.

2. Pressure Drop Analysis (Task 6.1) - F. T. Matthews

An analysis was carried out to determine the effect of temperature on primary loop operating point and control rod pressure drop.

A decrease in coolant temperature will increase both the coolant density and viscosity. The effect of decreasing the temperature will be to increase the system pressure losses and for a fixed characteristic performance curve of the circulating pump, the system flow rate will be reduced.

Figure 6-1 shows the temperature effect on primary system operating point and control rod pressure drop. Also plotted is a prorated pump operating line based on a pump selected to match 105% of the calculated loop head loss of 146.5 ft. H₂O at rated flow of 8990 gpm for a uniform flow core.

(Page F-2 of Report):

2. Pressure Drop Analysis (Task 6.1) - F. T. Matthews

An analysis was carried out to determine the effect of temperature on primary loop operating point and control rod pressure drop.

A decrease in coolant temperature will increase both the coolant density and viscosity. The change in density will cause the pump to draw more power, and tend to produce a slightly higher Reynolds Number. The predominant effect, however, will be the reduction in Reynolds Number, due to a great change in viscosity. This will cause higher friction factors, thus increasing system pressure loss. For a fixed characteristic performance curve of the circulating pump (temperature has little or no effect on this), the system flow rate will be reduced. The pressure drop acting on the control rods at temperatures substantially below operating level will then depend on the combined response, of the various parallel paths making up the second pass, to the higher friction factors, partly offset by the decreased flow rate.

Figure 6-1 shows the temperature effect on the primary system operating point and control rod (actually, overall second pass) pressure drop. Also plotted is a characteristic curve representative of the pump that is expected to be used for the SM-2 plant, based on a specification allowing for a 5% margin over the calculated required pressure drop of the primary loop.

At normal operating temperature Figure 6-1 shows that this specification will result in a 2-1/2% increase in rod pressure drop, while normal (80°F) or emergency (40°F) startup temperatures will cause it to increase 8.3 and 8.8% respectively over exact requirements at operating temperature.

For a normal startup temperature, the primary system flow rate will be 4% below the operating flow with a 12-1/4% increase in system pressure drop over the design point. At an emergency startup temperature of 40°F, the flow rate will be an additional 1-1/2% lower with a pressure loss of 15-1/2% above the design point.

Figure 6-1 also shows that over the greater range of temperatures, the control rod pressure drop, which is also the control rod static loading, remains fairly constant. Below 80°F the rod drop increases and is estimated to be 8% greater than the operating value at an emergency startup temperature of 40°F. The reference point is 30.1 ft. of water at 515°F, based on conversion to operating temperature of the test data at 80°F.

3. Structural Tests (Task 6.2) - F. Coleman

In order to ascertain the effect of cold-rolled fuel plates on outer fuel plate deflection characteristics, fuel element 26 (0.040" thick side plates and fuel plates containing depleted UO₂) was tested. The procedure was the same as followed with previous deflection testing. The results are shown in the accompanying graph, Fig. 6-2, which presents least square plots of the deflection-load relation for various other fuel elements in comparison with this element 26. The dashed line represents a derived deflection-load characteristic for an element "E" with 0.040" depleted fuel plates and side plates. This line was obtained by directly applying the ratio of slopes of the deflection curves of the annealed solid and annealed depleted 0.030" fuel plates to the results of the annealed solid 0.040" fuel plates.

By comparing the plots of element "E" and 26 it is noted that a cold-rolled element offers 18% greater resistance to deflection than does a similar element with annealed plates at the same pressure loading.

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

During this period, final modifications of the test rig have been made and tests of two types of fuel plates have been performed.

a. Test of a Brazed Fuel Plate

One of the fuel plates tested was of brazed construction and was cut from a dummy SM-1 element. The plate was subjected to transverse temperature profiles shown in Fig. 6-3. Resulting plate deflections are shown in Fig. 6-4. A comparison of the initial or zero temperature deflection of the test plate with measurements of the surface of the outer fuel plate of a production brazed SM-1 Core II element is given in Fig. 6-5. The agreement is quite good.

b. Test of a TIG Welded, Depleted UO₂ Core, Cold Rolled Fuel Plate

The second fuel plate tested was cut from a TIG welded fuel element of the reference SM-2 design. The plate contained a core of depleted UO₂. Temperature profiles induced on this plate are shown in Fig. 6-6 and the resulting plate distortion at the 89° temperature differential shown in Fig. 6-7. This differential was plotted since it closely represents the expected reactor operating differential (See Task 2, APAE Memo 197, for calculated plate temperature profile). The comparison of test plate flatness with element fuel plate measurements is also given in Fig. 6-5. The reasons for the distortion of the test plate are thought to be release of slight residual welding stresses and/or machining stresses introduced during the cutting of the plate from the element. It is expected that a single plate, if it could be made to retain the isothermal flatness measured while part of an element, would show significant column strength against buckling. Therefore, it is felt that the test results of the distorted test plate are inconclusive so far as SM-2 welded fuel elements are concerned.

A complete report of all thermal stress testing to date is being prepared and will be issued in the next report period.

Effort is concurrently being devoted to planning and designing an addition to the test rig which will enforce as-built flatness on the edges of the side plate portions of the test specimen.

5. Vibration and Collapse Testing (Task 6.3) - J. A. Christenson

Vibration testing of SM-2 fuel elements has been completed. Test results show no evidence of significant plate vibrations at element flow rates up to 630 gpm, 164% of the design element flow (384 gpm). Two welded fuel elements were tested.

The first vibration testing was done on a resistance welded, solid stainless steel plate element. Flow through the element was varied in 90 gpm increments from 180 gpm to 360 gpm (when this flow rate was reached the pickups shorted out). Slight plate movements of approximately 0.001" were observed. Frequency of these movements was less than 5 cps and quite random.

Since the movements indicated were so slight, and non-uniform, it is suspected that they may have been stray voltages from some external source. In any event, even if the movements exist they are not considered significant.

During the above testing, velocity probes were located 1/2" from the exit of each coolant passage. Measurements from these probes indicated that one channel of the element collapsed at between 540 and 630 gpm element flow.

Subsequent examination of the element revealed that during the installation of the vibration instruments one of the plates forming the collapsed channel had been distorted beyond dimensional tolerance. (Minimum channel width is 0.113", whereas the collapsed channel measured 0.111" before testing, with measurement taken 1-1/2" downstream of the entrance.) For this reason it is possible that the element would have withstood the 630 gpm if the channel had been within tolerance. However, collapse would have been imminent.

The second element tested, which was TIG welded and the fuel plates, annealed, contained a depleted UO_2 core. Collapse testing was performed prior to vibration instrumentation to avoid the difficulty experienced during the first test.

In this test the depleted core element also showed collapse between 540 and 630 gpm element flow. Channel measurements before testing proved the collapsed channel to be well within dimensional tolerance (0.119") although slightly below average.

A plot of channel velocities at 540 gpm element flow (Fig. 6-8) shows the collapsed channel to have the highest velocity in the element. The fact that the outer channels show extremely low flow is considered of no consequence since the outer fuel plates were subjected to pressure differentials roughly 10 times normal. A plot of collapsed channel velocity versus average element flow velocity is given in Fig. 6-9. The velocity of the adjacent channel is presented to show its velocity increase resulting from the collapse.

After completion of this test, the same element was fitted with an entrance comb and the test rerun. This time no collapse occurred at the 630 gpm flow rate. Based on this information, an entrance comb has been recommended for the SM-2 fuel element design and has been incorporated in the latest fuel element drawings.

Vibration testing of this element gave essentially the same results as the first element tested. Prior to testing the instruments were re-wired with special care being taken to insure reliability. Even with this extra care, the design carrier frequency of 20 KC could not be used successfully and had to be reduced to 2KC to avoid short circuits. At 2 KC carrier frequency, satisfactory operation was obtained.

Vibration traces again showed low amplitude, low frequency movements, but again they may have been stray external voltages and are not considered significant.

The traces also showed some flow-induced plate deflection. At design flow rate (384 gpm) these deflections measured 0.003 to 0.013" but since the element did not have the benefit of combs during this test, and the deflections

were measured 1/8" from the leading and trailing edges of the plates, the deflections are not considered serious.

Conclusions:

1. Fuel plate vibrations are practically non-existent in SM-2 welded fuel elements.
 2. Flow induced channel collapse, a potential problem, has been eliminated by the incorporation of leading (and trailing) edge combs in the fuel element design.
 3. The presence of these combs will also insure vibration free operation of the fuel plates.
 4. Low magnitude fuel plate deflections at leading and trailing edges of fuel plates are not considered serious since they will also be eliminated by the use of the combs.
6. Single Element Flow Testing (Task 6. 4a) - J. A. Christenson

a. Stationary Elements

Single element flow testing of stationary fuel elements has the following basic objectives:

1. Experimentally determine the coolant flow distribution within the element for various core support designs. Distribution tolerance for the SM-2 has been set at $\pm 6\%$ of element average.
2. Experimentally determine static pressure drops across the stationary element for various core support designs. The objective of first pass design is minimum pressure drop. Second pass pressure drop must match control rod pressure drop (37.3 feet of 80° water at 363 gpm design flow rate).

During this report period, three combinations of bottom and top core support plates have been tested.

1. Bottom plate: Grid type (see Style 6, AEL-393, Task 3, APAE Memo 160).

Top plate: Grid Type

This design was tested because it was thought to have the best possible change of meeting the distribution requirement above. Test results show the

lattice flow to be approximately 5 to 10% below element average. Flow distribution within the element was between +3% and - 5.5% of average at design flow rate. A plot of channel velocities is given in Fig. 6-10. Overall pressure drop observed (see next paragraph) at this condition, was 19.7 ft. of water. (Water temperature during this and subsequent testing was approximately 80°F.)

2. Bottom plate: AEL-523 (See Task 3.0)

Top Plate: Grid Style above.

This combination of support plates was tested to evaluate the performance of the bottom plate with the least possible influence by the top plate. Velocity distribution was within +9% and -6.5% of average. Velocities are plotted in Fig. 6-11. Overall pressure drop at 350 gpm of 80° water was 17.9 ft. of water. Prior to testing this combination, the entrance to the test section was modified to provide more accurate pressure drop information. The original design caused a contraction pressure loss to be included in the overall drop which was not representative of reactor conditions.

Inspection of the test element and instrumentation after disassembly of the rig revealed that the lattice passages had become distorted due to numerous assemblies and disassemblies. Consequently, lattice flow measurements were below true values. A new element and instrumentation were used for the next test. The new instrument supports provide more accurate positioning of the velocity and static pressure probes, and control of lattice passage width.

3. Bottom plate: AEL-523 (See Task 3.0)

Top Plate: AEL-526 (See Task 3.0)

This combination is the latest design of top and bottom support plates. The top plate section tested represented the design found in the second pass region. Overall pressure drop through this combination was 41.4 ft. of water, 26.6 ft. of this occurring across the top support plate. This pressure drop is obviously excessive, as it would result in the stationary elements having a higher pressure drop than the control rods of rated flow. The flow passages of this "Style 8b" upper support structure, as finally designed, were actually much more restrictive than the original concept which had been evaluated analytically. The final design was the result of a whole progression of small modifications for structural reasons, or to keep the elements interchangeable, or to make their ends interchangeable.

In flow distribution, lattice flow was noticeably better than for any other combination tested to date. A plot of channel distribution is given in Fig. 6-12. The extremely high pressure drop across the top support plate caused damage

to the velocity measuring instruments. As a result, no data is available for several channels.

b. Control Rod Assembly

Three tests have been conducted on the SM-2 control rod.

1. As Designed.

The control rod as designed (AEL-445, Task 3.0, APAE Memo No. 197) showed outer channel flow deficiency of about 20% at the design flow rate of 363 gpm. Static pressure drop at this flow rate was 37.3 feet of water at 80°F.

2. Entrance Modification

To force more flow to the outer channels, the semi-circular cutouts on the front of the two long fuel plates were replaced. This raised the velocity in the outer channels slightly and lowered the center channel velocity. There was no measurable change in static pressure drop from the as-designed test.

3. Absorber Removal

To determine the effect of the absorber section on outer channel fuel element flow, a test was run with the absorber section removed. The outer channels now measured approximately average element velocity. In all control rod tests the "average" channel velocity is subject to question since not every channel was instrumented. Additional instrumentation is being installed in the control rod at the present time to provide more complete distribution information.

c. Confirmation Testing

To determine experimentally and conclusively the static pressure drop across the stationary fuel elements, a test section has been designed for the Task 11 test loop. The significance is that testing in this section will be done at design temperature and pressure. This test section is shown in Dwg. R9-50 1083.

Pressure drops to be measured are (1) across the bottom support plate (probes 1 and 2); (2) across the fuel element (probes 2 and 3); and (3) across the top support plate (probes 3 and 4). Probe 5 is a velocity probe to determine element flow rate.

Tests will be conducted on both first and second pass support plate design. In the design of first pass supports it is desirable to have as low a pressure drop as possible, while the second pass fixed element pressure drop must match that of the control rods.

7. Full Scale Water Rig (Task 6.4b) - D. England

A decision to perform the full scale flow test in water was received from the A. E. C. on November 6, 1959. Accordingly, all final design work is proceeding with detailed design scheduled from completion on February 1, 1960. Completion of the entire test, exclusive of final reports, is anticipated by the end of September 1960.

Loop, vessel and core support construction will consist of a coated carbon steel assembly to ensure water purity. Fuel element construction will be of solid stainless steel. All sizes and surfaces have been geometrically reproduced at full scale. The rig is being designed for a rated flow of 9000 gpm at a temperature of 200°F.

Core flow balancing such as required will be done by measuring and adjusting individual element channel velocities by element discharge orifices, rather than by measurement of gross element flow as done in past flow work. This approach is necessary because of the multi-discharge hole type of top core plate being considered and the lack of discrete end boxes which provided positive continuity of flow.

EFFECT OF TEMPERATURE ON PRIMARY LOOP FLOW AND CONTROL ROD PRESSURE LOSS

FIG 6-1

FIGURE 1 EFFECT OF TEMPERATURE ON PRIMARY LOOP FLOW AND CONTROL ROD PRESSURE LOSS.

DEFLECTION-LOAD RELATION OF VARIOUS ELEMENTS SHOWING THE EFFECT OF COLD-ROLLED VS ANNEALED FUEL PLATES

FIG. 6-2

AXIAL LOCATION: CENTERLINE

SIDE PLATE THICKNESS = 0.050"

FIG. 6-3

622 14

COMPARISONS OF TEST PLATE FLATNESS WITH OUTER FUEL PLATE FLATNESS WHILE PART OF FUEL ELEMENT

SM-I CORE II ASSEMBLY
BRAZED PLATES

FIG. 6-5

Temperature Distribution Across Width of Test Plate

Axial location: Centerline

PLATES COLD ROLLED AND DEPLETED

FIG. 6-6

CHANNEL VELOCITY DISTRIBUTION PRIOR TO CHANNEL COLLAPSE

NOMINAL ELEMENT FLOW: 540 g.p.m.

CHANNEL 'F' COLLAPSED AT NEXT HIGHER FLOW RATE (630 g.p.m.)

FIG. 6-8

PLOT OF CHANNEL VELOCITIES SHOWING COLLAPSE OF CHANNEL F

FIG 6-9

STATIONARY ELEMENT FLOW DISTRIBUTION

BOTTOM CORE SUPPORT PLATE: AEL-393 (STYLE 6)

TOP CORE SUPPORT PLATE: AEL-393 (STYLE 6)

NOMINAL ELEMENT FLOW: 350 g.p.m.

FIG 6-10

STATIONARY ELEMENT FLOW DISTRIBUTION

BOTTOM CORE SUPPORT PLATE: AEL-523

TOP CORE SUPPORT PLATE: AEL-393

NOMINAL ELEMENT FLOW: 350 g.p.m.

FIG. 6-11

STATIONARY ELEMENT FLOW DISTRIBUTION

BOTTOM CORE SUPPORT PLATE: AEL-523 (STYLE 9)
TOP CORE SUPPORT PLATE: AEL-526 (STYLE 8B)

NOMINAL ELEMENT FLOW: 350 g.p.m.

(NO DATA COLLECTED FOR
CHANNELS B, G, I, L)

FIG. 6-12

18	1	MALE CONNECTOR
17	8	1/2" NPT VALVE
16	1	FLUG (EXIT)
15	1	1/2" NPT VALVE
14	1	1/2" NPT VALVE
13	1	1/2" NPT VALVE
12	1	1/2" NPT VALVE
11	1	1/2" NPT VALVE
10	1	1/2" NPT VALVE
9	1	1/2" NPT VALVE
8	1	1/2" NPT VALVE
7	1	1/2" NPT VALVE
6	1	1/2" NPT VALVE
5	1	1/2" NPT VALVE
4	1	1/2" NPT VALVE
3	1	1/2" NPT VALVE
2	1	1/2" NPT VALVE
1	1	1/2" NPT VALVE

ALCO PRODUCTS, INC.
ANN ARBOR, MICH.
ENGINEERING DEPT.
R9-50-1083

FLOW TEST SECTION
SM-2 TEST LOOP
R9-50-1083

G. CRITICAL EXPERIMENT PROGRAM

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

During this report period mockup experiments of the SM-2 and SM-1 cold clean and the SM-2 midlife cores were completed. The final report has been written and is being printed.

Approximately 99% of the program effort has been completed and 98% of the approved funds expended, or committed.

2. Final Core Mockup (Task 7.7)

a. The Final SM-2 Mockup - S. H. Weiss

The final SM-2 mockup core composition and seven rod bank critical positions are tabulated in Table 7-1.

TABLE 7-1
SM-2 COLD CLEAN CORE COMPOSITION

	<u>Stationary Elements</u>	<u>C. R. Elements</u>	<u>Total</u>
Boron Loading	53.0 gm B ¹⁰	7.94 gm B ¹⁰	60.9 gm B ¹⁰
Fuel Loading	31669.20 gm U ²³⁵	4730.88 gm U ²³⁵	36400.08 gm U ²³⁵
Side Plate Thickness	0.0327 in.	0.0306 in.	---
Seven rod critical bank position for water reflected case			7.142 in.
Seven rod critical bank position for steel reflected case*			6.974 in.

* Steel reflector consisted of four (4) steel corner reflectors plus 2-1/2" of steel along the reactor sides and laminated with water in the following manner; 1/2" steel, 1/4" H₂O, 1" steel, 1/4" H₂O, 1" steel.

It was not feasible to directly measure the temperature coefficients of the SM-2 over the temperature range desired (room temperature to 510°F); so two separate experimental methods were used. For a temperature range from 68 to 155°F, the temperature coefficient, at normal atmospheric pressure, was determined by heating the water in the reactor tank. In order to determine the temperature coefficient above 155°F, aluminum strips were used to displace the water and thus simulate moderator temperatures up to 510°F. The measurements were made with a laminated steel reflector around the core. Figure 7.1 shows equivalent temperature at 2000 psi vs. negative ΔK_E in cents.

The temperature coefficient was determined by the slope of the aluminum curve in Fig. 7.1 for temperatures above 130°F. This method does not take into account the absorption of neutrons in the aluminum or the imperfect distribution of aluminum in the core. These effects may be neglected with little loss of confidence in the data.

The negative ΔK_E due to increasing the core temperature, at 2000 psi, from 103.5 to 510.2°F, and the reflector temperature from 103.5 to 477°F, measured by displacing water with aluminum, is 889.7 cents.

TABLE 7.2
TEMPERATURE COEFFICIENTS

Temp. °F at 2000 psi	7 Rod Bank Position Inches	Temp. Coefficient cents/°F
150	7.150	-1.15
370	8.850	-2.50
510	11.291	-5.20

The boron and uranium reactivity coefficients were measured by differences on calibrated control rods as a consequence of substitution of known increments of boron or uranium into respective element positions in one reactor quadrant. Core symmetry was assumed in order to obtain the average coefficients for the entire core.

TABLE 7.3
REACTIVITY COEFFICIENTS OF URANIUM-235 AND BORON-10

<u>Element Position</u>	<u>U²³⁵ Worth in cents/gram</u>	<u>B¹⁰ Worth in cents/gram</u>
45	0.312	87.5
46	0.220	58.9
47	0.098	25.5
54	0.321	87.7
55	0.295	79.5
57	0.084	22.6
65	0.211	57.0
66	0.146	38.6
67	0.057	14.6
74	0.110	28.0
75	0.098	25.6
76	0.067	17.0

The average B¹⁰ worth in the SM-2 core is 42.54 cents per gram.

The average U²³⁵ worth in the SM-2 core is 0.157 cents/gram.

During the final mockup experiments, more complete reflector measurements were made. Since the maximum steel available allows for only 2-1/2" of steel on all four sides, data were obtained for both four sides-four corner reflection, and two adjacent sides-two corner reflection. Fig. 7.2 describes the reactivity effects of the design lamination of steel and water reflectors and solid steel corner reflectors. The preliminary mockup reflector points are also plotted in Fig. 7.2. Extrapolation of the four-side-four cornered laminated reflector curve indicates a reflector worth in the thick reflector region not exceeding +85.0 cents in the SM-2 final cold clean mockup.

The effect of rotating the two outer rows of elements 90° so that the fuel plates in the outer rows are normal to the reflector was found to be worth 8.91 cents negative.

To determine the effect of substituting stationary elements for control rods, the standard SM-1 element was placed in position #72. Critical six rod bank positions were taken and calibrations made with rods A, B, and C fully withdrawn in turn. Element #72, with its normal uranium and boron loading, was then substituted for rods A, B, and C respectively, and new critical positions and calibrations determined. The substitution element differed from the control rods, having 157.6 gm more U²³⁵ and 0.34 gm more B¹⁰. Table 7.3 shows the results of these measurements.

TABLE 7.4
EFFECT OF SUBSTITUTING STATIONARY FUEL ELEMENTS

	6 Rod Bank Critical Position	6 Rod Bank Worth cents/in.	Worth Cents
Rod A withdrawn	5.914"	209 @ 5.945"	+37.6
#72 substitution in Rod A position	5.733"	206 @ 5.767"	
Rod B withdrawn	6.175"	239 @ 6.207"	+36.0
#72 substitution in Rod B position	6.022"	231 @ 6.061"	
Rod C withdrawn	5.720"	217.5 @ 5.775"	+43.6
#72 substitution in Rod C position	5.520"	219.0 @ 5.560"	

b. Final SM-2 Midlife Mockup - E.W. Schrader

Radial burnout of nuclear fuel and poison in the SM-2 midlife core were mocked up by dividing the core into several concentric rings about the core axis, and loading the fuel elements in each ring with the proper amount of fuel and nuclear poison to mockup the calculated midlife core loading. Measurements on the SM-2 midlife core mockup include seven rod bank position at criticality, rod calibration, and stuck rod conditions. The midlife core with a water reflector was critical after withdrawal of the seven rod bank to 4.946 inches. The seven rod bank worth is 341 cents/inch at 4.975 inches withdrawal. The stuck rod conditions are summarized in Table 7.5.

TABLE 7.5
SM-2 (FINAL MOCKUP) MIDLIFE CRITICAL ROD CONFIGURATION
AT 68° WITH A WATER REFLECTOR

Case	A	B	C	D	E	F	G	Two Rod Bank Worth
1	0	0	10.669	10.669	0	0	0	116 cents/in. @ 10.759 in.
2	0	0	12.190	0	0	0	12.193	91 cents/in. @ 12.297 in.
3	0	0	0	11.344	0	0	11.342	107 cents/in. @ 11.423 in.
4	0	12.037	0	0	0	0	12.037	113 cents/in. @ 12.145 in.
5	0	16.885	0	0	0	16.888	0	34.5 cents/in. @ 17.200 in.
6	0	0	0	14.726	0	14.727	0	53.5 cents/in. @ 14.898 in.
7	14.692	0	0	14.691	0	0	0	53.1 cents/in. @ 14.847 in.

The total core loading for the final SM-2 midlife mockup consisted of 29190.24 gm U^{235} and 20.39 gm B^{10} . Side plate thicknesses were 0.0327 in. and 0.0306 in. for stationary and control rod elements, respectively.

c. SM-1 Mockup -- R. A. Robinson

Measurements of heterogeneity effects introduced by mocking up the SM-1 core with SM-2 fuel plates were accomplished using a fuel grouping technique. The plastic separators on the side plates of one element were modified so that the fuel plates and stainless steel clad used in obtaining the SM-1 metal to water ratio could be assembled in discrete bundles. The eleven fuel plates per element which mocked-up the fuel loading of the SM-1 were first assembled into one discrete bundle, along with the stainless steel clad, in the center of the element. They were then assembled into two discrete bundles, three discrete bundles, etc., until the maximum of eleven bundles was reached.

Fig. 7.3 shows the curve of reactivity change versus number of bundles, obtained from these measurements. Since the curve is essentially flat between 9 and 11 bundles, it seems reasonable to linearly extrapolate it to 18 bundles, the actual SM-1 fuel element configuration. The heterogeneity worth thus obtained was 0.75 cents for element #22 in which the measurements were made. The average heterogeneity worth was 0.60 cents per stationary element, obtained by dividing the value determined for element #22 by the ratio of the local to average uranium worth.

As previously published*, the final SM-1 mockup core was loaded with 22.310 kg U^{235} and 15.2 gm B^{10} . Correcting the mockup data to the actual SM-1 conditions indicates a B^{10} loss in the actual SM-1 Core of $22.0\% \pm 1.9\%$

* APAE Memo 223, SM-2 Reactor Core and Vessel Review Report, May 28, 1959 to August 24, 1959.

FIGURE 7-1

AK_E vs. EQUIVALENT TEMPERATURE AT 2000 PSI

- Heating Points
- △ Aluminum Points
- ▽ Aluminum Points (stationary elements only)

FIGURE 7-2

ΔK_t vs LAMINATED STEEL REFLECTOR THICKNESS
COLD CLEAN SM-2 MOCKUP

FIGURE 7-3

ΔK_E vs. NUMBER OF FUEL BUNDLES

H. CONTROL ROD DRIVE DEVELOPMENT*

1. Progress (Task 11.0) - G.M. Van de Mark

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

All components of the prototype drive have been purchased, and approximately 25% of these items have been received. It is expected that nearly 60% of the purchased parts will have been shipped by the week of December 14, and the balance by the week of December 28. The prototype seal assembly has been received from Alco's Auburn Plant and is at present in the laboratory.

All material required for the fabrication of the test loop has been received with the exception of the test autoclave. Delivery of the autoclave from Alco's Dunkirk Plant has been promised for the week of December 28.

At the present time the test loop erection is approximately 30% complete. All support structures are nearly completed, the pump motor control and heater control panel has been completed and the surge tank, make-up pump, high and low pressure demineralizers, economizer, coolers, and de-oxygenator have been installed. It is expected at this time that the test loop will be completed during the third week in January.

As a result of the large expected static pressure drop across the SM-2 control rod, the possibility of balancing this pressure hydraulically has been investigated. From the point of view of designing a control rod drive mechanism, the hydraulic balanced control rod appears very feasible. The increased fabrication problems of vessel and core support structure, by the addition of transfer tubes and lower diaphragm, are discussed in Task 3.0

2. Improved Accessibility of Drive Components (Task 11.1) - G.M. Van de Mark

No further work has been done on this phase of Task 11 since the previous review period.

3. Seal Leakage (Task 11.2) - G.M. Van de Mark

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

4. Acceleration Device Design (Task 11.3) - G. M. Van de Mark

No further design 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. Van de Mark

The clutch tests indicate that the Maxitorq clutch is a much more desirable clutch than the Stearns. Due to the delay in obtaining clutch data, the Stearns clutch is being used in the prototype drive mechanism being fabricated. The feasibility of incorporating the Maxitorq clutch in the present clutch housing has been investigated, and it appears that major modification will be required to make this change. It is planned to modify the clutch assembly design incorporating the Maxitorq clutch and proper accelerating device (negator spring) when the final flow and pressure drop have been determined.

An APAE Memo documenting the clutch tests is being prepared, and it is expected to be completed by January.