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PRELIMINARY TECHNICAL REPORT
SM-1 CORE III
WITH TYPE 3 ELEMENTS

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ABSTRACT

This preliminary technical report covers design of the SM Type 3 element, its use in SM-1 Core III, and planned use in the Type 3 Core for PM-2A. The Type 3 element characteristics are compared with Type 1 (SM-1 Core I) element nuclear, metallurgical, thermal and hydraulic characteristics and fabrication. The effect of using the Type 3 element in SM-1 Core III and its planned use in a Type 3 Core for PM-2A is discussed with regard to operation, shielding, reactor safety and all conceivable special problems.

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1.0 SUMMARY

SM-1 Core III will be used as a test for Type 3 elements in a PM-2A core. The test will consist of operating a core of 37 Type 3 elements in the SM-1 plant at Ft. Belvoir to demonstrate the satisfactory thermal and nuclear performance of a reactor containing Type 3 elements and demonstrate that the 37 element Type 3 core can be shut down with its most reactive rod stuck in the full out position at the most reactive time in core life.

The use of the Type 3 element offers a significant gain in lifetime, accomplished by increasing the loading of fuel and burnable poison in the thicker fuel plate of the Type 3 element.

Metallurgical analysis and test indicate that the Type 3 fuel element will be capable of withstanding the maximum expected burnup and that successful performance of Type 3 elements in SM-1 Core III and in PM-2A can be anticipated.

SM-1 Core III will be as nearly like the Type 3 core in PM-2A as possible. The plate fuel and burnable poison loading, the operating power level, the number of elements, and the arrangement of flux suppressors will be the same for both cores.

The stuck rod problem at the most reactive time in life was investigated with survey-type calculations for SM-1 Core III and the PM-2A with Type 3 elements. On the basis of these calculations the PM-2A with Type 3 elements will have an estimated shutdown margin of approximately 1.3% ρ . A more detailed study of this problem is recommended.

A thermal analysis was performed on the use of Type 3 elements in a 37-element core. Steady state analysis indicates that the core can operate safely during design and scram power conditions. Local nucleate boiling occurs at design conditions and bulk boiling occurs at scram power conditions. Transient analysis showed that during the critical period following a loss of flow accident, the minimum DNBR is well above the minimum allowable of 1.5.

Dose rates in working areas during operation of SM-1 Core III and after shutdown are not expected to be different from doses reported for SM-1 Core I. Dose rates from irradiated Type 3 elements will have little effect on refueling operations.

A study of various cores and their effects on the fast nvt experienced by pressure vessels indicates that substitution of Type 3 elements in cores for the SM-1 will not cause an appreciable increase in the neutron flux above 1 Mev at the wall of the reactor vessel.

A study is recommended to determine whether a redesign of the spent fuel pit is necessary.

The nuclear parameters that affect reactor kinetics are not radically modified by the substitution of Type 3 elements; therefore, the energy release of the nuclear excursion postulated in the hazards summary report for the SM-1 would be of the same order of magnitude for SM-1 Core III. The maximum credible accident was defined as a series of failures occurring simultaneously in the primary and secondary systems which result in a release of all stored energies in these systems to the vapor container. These thermal energy releases are much greater than the energy releases calculated for any of the postulated nuclear excursions considered for this plant. It is concluded that the use of Type 3 elements for the SM-1 will have no significant effect on the severity of the maximum credible accident postulated in the hazards summary report.

The same areas were also considered for the PM-2A, resulting in the same conditions for that system.

2.0 INTRODUCTION

SM-1 Core III is to be a 37-element core as a test of a Type 3 core for use in PM-2A. Since SM-1 Core III is to be a prototype core, it is necessary to discuss both Type 3 elements in PM-2A as well as in SM-1 Core III. A core approximating the PM-2A will be made up in the SM-1 for an initial testing.

The SM-1 core normally contains 45 elements. During operation with the 37-element SM-1 Core III, the eight extra fuel element positions will be filled with dummy elements containing flow control devices. These element positions will be defined later in this report.

Two Type 3 test elements which have operated in SM-1 Core II (SM-2A and SM-2B) for the purpose of this analysis will remain in SM-1 Core III.

The Type 3 element has a thicker fuel plate than the Type 1 element. The Type 3 element is a welded assembly, whereas the Type 1 element is a brazed assembly. Other features of these elements are listed in Table 2.1 below. Additional technical information on the Type 1 element is presented in the literature.⁽¹⁾

The SM-1 Type 3 fuel element assembly (Dwg. R9-13-1052) is identical to PM-2A Type 3 fuel element assembly except that the SM-1 element overall length is 33-5/8 in. compared to 31-5/8 in. for the PM-2A element, since the SM-1 accommodates a hold-down spring. The Type 3 control rod fuel element is shown in Dwg. D9-13-1047.

This report was prepared under Subtask 3.8 of the FY 1962 Program for Engineering Support and Development of Army Pressurized Water Reactor Power Plants.*

* AP Note 378, September 6, 1961.

TABLE 2.1
CHARACTERISTICS OF TYPE 1 AND TYPE 3 ELEMENTS

<u>Fuel Elements (Fixed)</u>	<u>Units</u>	<u>Type 1</u>	<u>Type 3 for SM-1 Core III</u>
Outside dimensions	(in.)	2.863 x 2.844 x 27	2.863 x 2.844 x 27
Weight of U-235 per element	(gm)	515.16	833.4
Weight of UO ₂ per element	(gm)	630.36	1021.8
Weight of B-10 per element	(gm)	0.4646	0.8964 (est)
<u>Fuel Elements (Control Rod)</u>			
Outside dimensions	(in.)	2.624 x 2.619 x 24.8125	2.624 x 2.619 x 24.8125
Weight of U-235 per element	(gm)	417.76	655.68
Weight UO ₂ per element	(gm)	512.16	803.95
Weight of B-10 per element	(gm)	0.3771	0.7056 (est)
Enrichment of U-235	(%)	93 ± 1	93 ± 1
<u>Fuel Plates (Fixed)</u>			
Type of fuel		UO ₂ (Geneva)	UO ₂ (spherical)
No. plates/element		18	18
Active length		21.75	22
Size of fuel plates	(in.)	0.030 x 2.778 x 23	0.040 x 2.589 x 23.5
Type of cladding		304L SS	347
Wall thickness of cladding	(in.)	0.005	0.005
Meat thickness	(in.)	0.020	0.030
Matrix material		302 LB	347 SS
Weight of U-235 per plate	(gm)	28.62	46.3
Weight of UO ₂ per plate	(gm)	35.02	56.8
Weight of B-10 per plate	(gm)	0.02581	0.0498 (est)
<u>Fuel Plates (Control Rod)</u>			
Type of fuel		UO ₂ (Geneva)	UO ₂ (spherical)
No. of plates/element		16	16
Active length		21.125	21
Size of plates	(in.)	0.030 x 2.558 x 23	0.040 x 2.589 x 23.5
Type of encasement		304L SS	347 SS
Wall thickness of cladding	(in.)	0.005	0.005
Meat width	(in.)	2.281 min	2.400 min
Meat thickness	(in.)	0.020	0.030
Weight of U-235 per plate	(gm)	26.11	40.98
Weight of UO ₂ per plate	(gm)	32.01	50.24
Weight of B-10 per plate	(gm)	0.02357	0.0441 (est)

DATE	LET	REVISION	RECORD	AUTH	DR	CHK
1-5-62	A	REMOVED DETAILS OF NO. 1 AND NO. 2; SEPARATE NO. 1 AND NO. 2				
		ADDED REF TO PCD				
		REMOVED LOW IMPURITY				
		STYLE WAS: SM-1 FUEL ELEMENT (EUTRUM FOR SM-1, SM-1A & PLATE REACTOR) EIC-5188				
1-23-62	B	ADDED NOTE FOR TIGHTNESS OF HANDLE				
		ADDED FINISHED DIMENSIONS ON HANDLE EIC-5189				
2-20-62	C	SM TYPE 3 CORE AND FUEL ELEMENT NO. 4 FABRIC. SYMBOL TO BE ADDED EIC-5197

FUEL ELEMENT NO. 4 FABRICATOR'S SYMBOL TO BE MARKED ON HANDLE END OF SIDE PLATE WITH A BURGESS VIBRATOR TOOL OR EQUIVALENT.

QTY	NO.	DESCRIPTION
6	1	B9-13-2195 COMB
5	1	A3-13-2077 PIN - HANDLE
4	2	A3-13-2193 HANDLE
3	2	B9-13-2194 PLATE EXTENSION
2	2	C9-13-2214 SIDE PLATE
1	16	B9-13-2213 FUEL PLATE
P.C.T. TENS. NO. 103		
MATERIAL		
DESCRIPTION		
ALCO		
ALCO PRODUCTS, INC.		
NUCLEAR POWER ENGINEERING DEPT.		
SCHENECTADY, N. Y. U. S. A.		
DATE FULL 1/2/61		
MATERIAL SPEC		
CHK		
APPN		
NET		
FUEL ELEMENT (CONT ROD)		
PART NO. D9-13-1047

3.0 CORE ANALYSIS

3.1 INTRODUCTION

SM-1 Core III, shown in Fig. 3.1, will be a prototype test for Type 3 elements in the PM-2A core configuration, shown in Fig. 3.2. The SM-1 Core I configuration is shown in Fig. 3.3 for comparison. For the prototype test, a 37-element core of Type 3 elements (SM-1 Core III) will be operated in the SM-1 plant at Fort Belvoir. The purpose of the prototype test, in addition to supplying a core for the SM-1, is to demonstrate the satisfactory thermal and nuclear performance of Type 3 elements in the PM-2A plant. Operational data from SM-1 Core III will provide a realistic test of design models for the Type 3 core in PM-2A. The prototype tests will also permit verification that Type 3 elements in PM-2A can meet a one rod stuck out criteria. SM-1 Core III will be operated with rods A and B in the fully withdrawn position; they will be available as an additional means of shutdown.

SM-1 Core III will be as nearly like the final design of the Type 3 core in PM-2A as possible. The characteristics of these two cores are compared with each other and with SM-1 Core I in Table 3.1. From the nuclear standpoint, there are three major differences between SM-1 Core III and the Type 3 core in PM-2A:

1. In Core III, two control rod fuel elements and associated control rods A and B will occupy element positions 33 and 55 respectively. Control rods A and B will remain fully withdrawn and provide additional shutdown protection. In the Type 3 core for PM-2A, these element positions will be occupied by stationary fuel elements.
2. SM-1 will operate at 440°F, PM-2A at 509°F.
3. Two high burnup test elements, SM-2A and SM-2B, will be included in Core III element positions 37 and 51 respectively.

3.2 NUCLEAR CHARACTERISTICS OF TYPE 3 ELEMENTS

3.2.1 Initial Fuel and Burnable Poison Loadings

3.2.1.1 Fuel Loading

The plate fuel loading of the Type 3 element is 46.30 gm U-235. The U-235 loading of Core III is estimated to be 29.2 kg.* Core III contains 28 stationary elements, two test elements and seven control rod fuel elements. The Type 3 core

* An uncertainty in the fuel depletion of SM-2A and SM-2B at the end of SM-1 Core II life results in an uncertainty in the initial fuel and poison inventory of Core III.

Figure 3.1. SM-1 Core III Configuration

Figure 3.2. PM-2A Type 3 Core Configuration

Figure 3.3. SM-1 Core I Configuration

Figure 3.4. Axial Power Distribution for PM-2A Core with Type 3 Elements with Various B-10 Loadings

TABLE 3.1
CORE PARAMETERS FOR SM-1 CORE I, SM-1 CORE III,
AND THE TYPE 3 CORE IN PM-2A

<u>Core Data</u>	<u>Units</u>	<u>SM-1 Core I</u>	<u>SM-1 Core III</u>	<u>Type 3 Core in PM-2A</u>
Thermal power	Mw	10	10	10
Power density (nominal steady state)	kw/liter	77	87	87
Average core diameter	in.	22.2	20.2	20.2
Height	in.	22	22	22
Core active volume	liters	139.5	114.7	114.7
No. fixed fuel assy's		38	30	32
No. control fuel assy's		7	7	5
Core loading, U-235	kg	22.50	29.2	29.95
B-10 in core	gm	15.75	30.6	32.2
U-235	%	93 $\pm$ 1	93 $\pm$ 1	93 $\pm$ 1
Initial reactivity, cold	%	15.4	10.9	12.1
Initial thermal neutron flux	n/cm ² -sec	1.4 x 10 ¹³	8.6 x 10 ¹² +	8.9 x 10 ¹² ++
Five rod bank worth (68°F)	% Δ k	19.0	22.1	22.1
Expected energy release	MWYR	16.4	~27*	~24*
Average core coolant temp	°F	440	440	509

* Due to the lower operating temperature of SM-1 Core III, its lifetime is approximately three MWYR greater than the PM-2A with Type 3 elements.

+ at 440°F

++ at 510°F

in PM-2A, with U-235 loading of 29.95 kg, contains 32 stationary elements and five control rod fuel elements. The difference in the loadings arises from the fact that the two cores contain different numbers of stationary and control rod fuel elements; each stationary element contains 18 plates, while each control rod fuel element contains only 16 plates. Another difference arises because the loading of the test elements differ from that of the other stationary elements.

3.2.1.2 Burnable Poison Loading

The poison loading of SM-1 Core III is determined by the poison loading of the Type 3 Core in PM-2A. The current best estimate of the boron loading of the Type 3 core in PM-2A is 32.2 gm or 49.8 mg per fixed element plate and 44.1 mg per control rod element plate. Based on these PM-2A core plate loadings, the estimated poison

loading of the SM-1 Core III, including the B-10 remaining in test elements SM-2A and SM-2B, is 30.6 gm.* These loadings are subject to change when the experimental results from the PM-2A Type 3 core mockup experiments and more accurate determinations of the shutdown margin at the most reactive time in life available.

It will be noted in the calculations that follow, several B-10 loadings have been used (e.g. 32.2, 32, 31.8 gm). This was due to the uncertainty in the loading of this core when the calculations were performed; however, all of these values are within the poison loading uncertainty of ± 5 percent.

The poison loading of the Type 3 core in PM-2A must be low enough to provide a sufficient startup margin to override maximum Xe and allow for loading tolerances and design model uncertainties. Conversely, it must not be so low that it will result in a critical control rod bank position deep enough in the core to cause an undesirable increase in the initial maximum axial power density. This latter effect, illustrated in Fig. 3.4 for two PM-2A boron loadings, is not serious for the Type 3 core in PM-2A due to the large thermal margin. The poison loading of the Type 3 core in PM-2A was determined to provide a k_{eff} of 1.03 at the beginning of life, at an operating temperature of 509°F and with equilibrium Xe at 10 Mw. The resulting initial excess reactivity of 2.91 percent ρ represents a compromise between a sufficient startup margin and the ability to meet the stuck rod shutdown requirement.

3.2.2 Shutdown Capability with One Rod Stuck Out

The Type 3 core in PM-2A is designed to meet a shutdown requirement with one rod stuck fully out when the core is at its most reactive time in life. The best current estimate of the shutdown margin in a cold, clean Type 3 core in PM-2A with its central rod fully withdrawn and at the time of maximum reactivity is 1.3 percent ρ . As may be observed in Fig. 3.5, the most reactive time in core life occurs at a U-235 burnup of approximately 16.5 percent.

These estimates were based on a uniform depletion study of the Type 3 core in PM-2A which contained 31.8 gm B-10 initially. The calculated results were normalized to ρ_{equal} to 4.41 percent at the beginning of life, which is the design excess reactivity for PM-2A with Type 3 elements with no Xe at 510°F. The reactivity increase at 16.5 percent U-235 burnup was estimated to be 1 percent ρ .

The SM-1 Core III is estimated to be less reactive at 0 MWYR than the PM-2A core by 1.2 percent ρ . Assuming this reactivity difference to remain until the most reactive time in life, SM-1 Core III should meet a one stuck rod criterion (with A and B fully withdrawn) with a 2.5 percent ρ margin.

* An uncertainty in the fuel depletion of SM-2A and SM-2B at the end of SM-1 Core II life results in an uncertainty in the initial fuel and poison inventory of Core III.

A useful comparison can be made between the Type 3 core in PM-2A and PM-2A Core II, for which experimental data has been obtained from a zero power experiment. In Table 3.2, the maximum reactivity of the Type 3 core in PM-2A is compared to that for PM-2A Core II. With the exception of the increase of the reactivity of the PM-2A with Type 3 elements with depletion (estimated to be 1 percent ρ on the basis of the uniform depletion studies mentioned above), all of the reported values are taken from preliminary results of a PM-2A Type 3 core mockup experiments or from the ZPE performed on PM-2A Core II. (2) Note that PM-2A Core II is slightly more reactive than the Type 3 core in PM-2A, and that the Type 3 core five rod bank worth determined experimentally appears to be slightly greater than that of Core II. Core II has an estimated shutdown margin of \$1.10 (0.85 percent ρ) initially with the most reactive rod fully withdrawn.

TABLE 3.2
REACTIVITY COMPARISON OF PM-2A CORE II
AND A TYPE 3 CORE IN PM-2A

	<u>PM-2A Core II</u>	<u>Type 3 Core in PM-2A</u>
Maximum excess reactivity	\$18.40 ^{1*}	\$18.10 **
Five rod bank worth	31.00 ^{3***}	31.90 ***
Four rod bank worth	- - - **	19.80 ***
Estimated shutdown margin, most reactive rod fully withdrawn	1.10 ***	1.70 (1.3% ρ)****

Notes

- * Cold, clean, from experiment⁽²⁾
- ** Cold, no Xe, with fission products, uniform burnup assumed, with 31.8 gm B-10 initially.
- *** Experimental values. Measurements were made with boron control rods rather than europium rods. The relative worths of boron and europium rods were measured in zero power experiments⁽³⁾ for SM-1 Core I. Substitution of a europium absorber for a boron absorber results in a worth addition of + 21.1 cents. Rod worth from the PM-2A Core II ZPE⁽²⁾
- **** Value estimated on basis of above data

Curve A: Includes fission products except Xe

Core Conditions:

31.8 gms B-10

Operating Temperature 509°F

Curve B: Includes fission products plus 10 Mw equilibrium Xe

Figure 3.5. Reactivity Vs Burnup for Type 3 Elements in PM-2A

Figure 3.6. Relative Axial Power Distribution (VALPROD) SM-1 Core III

The major uncertainties in the values presented in Table 3.2 and consequently in the estimate of the shutdown margin arise from:

1. The experimental uncertainty in the measured values
2. The effect on the reactivity increase of non-uniform depletion in the PM-2A core with Type 3 elements
3. The worth of the flux suppressors

The worth of the flux suppressors in the Type 3 core in PM-2A was estimated to be \$2.60 (2 percent ρ) on the basis of experiments performed in the SM-2 mockups.⁽⁴⁾

Additional uncertainties enter into the calculation of the reactivity increase of the Type 3 core in PM-2A. These arise from the uncertain boron content of Mylar tape used in the mockup experiments and in the uncertainty in the equivalence of internally and externally loaded boron. This latter point is further discussed in Section 3.3.2. The mockup experiments are used to determine the quantity of boron necessary to achieve a k_{eff} of 1.030 at 510°F with equilibrium Xe. If, after resolving these above mentioned uncertainties, the quantity of boron initially thought to be present is re-determined to be above 32.2 gm (with k_{eff} still 1.030), then the reactivity increase with depletion that would be calculated, would then be greater than 1 percent ρ . The shutdown margin with one rod stuck would hence be less than 1.3 percent ρ as estimated in Table 3.2.

The effect of non-uniform depletion on the reactivity increase with burnup of both cores will be investigated with the one-dimensional depletion code CANDLE. It is recommended that this effect also be investigated with the additional detail available with the use of the two-dimensional depletion code TURBO to further substantiate the value of the shutdown margin.

3.2.3 Average U-235 Depletion and Lifetime

Non-uniform depletion studies of a Type 3 core in PM-2A (using the IBM-704 code CANDLE⁽⁵⁾) gave an estimated energy release of 24 MWYR* and an average U-235 depletion of 42 percent. For these studies, the core contained 47 gm B-10. If the poison content is decreased to 32.2 gm, which is the value now estimated for a Type 3 core in PM-2A, the lifetime will be slightly increased (on the order of 0.5 MWYR). From these studies, it is estimated that SM-1 Core III will undergo an average U-235 depletion of 45 percent and will have a lifetime of approximately 27 MWYR. The reduced operating temperature of Core III, as compared to the PM-2A, results in the estimated increase of 3 MWYR in Core III lifetime.

* At 24 MWYR the core can no longer override 10 Mw equilibrium Xe

More accurate estimates of lifetime and depletion of both cores must await the completion of one and two-dimensional depletion studies. The one-dimensional depletion studies should be completed in April, 1962.

3.2.4 Power Distributions

Comparing the two cores, the most adverse power distribution will occur in SM-1 Core III. This is caused primarily by two factors:

1. The lower operating temperature of Core III results in more peaking in Core III than in the Type 3 core in PM-2A, due to an increase in moderator density.
2. In the SM-1 plant the edge of Core III is at a considerably further distance from its thermal shield than the edge of the core in the PM-2A is from its thermal shield. (7.268 in. in SM-1 as compared to 1.518 in. in PM-2A*). This results in increased power peaking at the Core III reflector interface due to the increased thermal flux at this point.

The most adverse power distribution in Core III occurs when core reactivity is at a maximum, since at this time the control rod bank is most deeply inserted into the core. Power peaking factors for each element are presented in Table 5.3 for Core III at 440°F. The relative axial power distribution is presented in Fig. 3.6. Correction factors for the calculated power distribution were obtained from detailed power distribution measurements in a PM-2A mockup containing an estimated 56 gm B-10.

3.3 CURRENT DESIGN PROBLEMS

3.3.1 Prediction of Nuclear Characteristic Variations with Depletion

Accurate prediction of the reactivity variation of the Type 3 core in the PM-2A with depletion is essential to determine the shutdown margin available with the most reactive rod stuck full out when core reactivity is at a maximum. None of the depletion code results for Type 3 elements have been verified by comparison with experiments**; therefore, the adequacy of the various models can be judged only by comparing predictions with experimental data from similar cores. It is anticipated, however, that the experimental data obtained during Core III depletion will serve as a realistic test of depletion models for Type 3 elements and will lend confidence to the prediction of PM-2A core characteristics using this element.

* Distance measured from core reflector interface at centerline of element 41 to inside edge of thermal shield along core radius.

** A comparison has been made for Type 1 elements in the SM-1 Core I. (6)

3.3.2 Comparison of Analytical Predictions with Experiments

Boron loaded Mylar tape is used in core mockup experiments to simulate internally loaded boron in the reference cores. These mockup experiments are used to test calculational models* and to determine the poison loadings. The analysis of these experiments have been hampered by two difficulties:

1. Determination of the boron content of the Mylar tape.
2. Evaluation of the reactivity equivalence of externally and internally loaded boron.

Due to the high reactivity worth of boron, it is essential that the quantity of boron present in a mockup core be accurately known. Lucius Pitkin has already performed an analysis on Mylar tape supplied by Alco, and arrangements have been completed to receive another analysis from the AEC's New Brunswick Laboratory.

The reactivity difference of the boron in the Mylar tape, as compared to boron loaded in a fuel plate was investigated by the use of one velocity transport theory. This reactivity difference is not accurately known, however, indications are that this effect is not great. Calculations including the multi-velocity effects are planned.

* (i.e., the comparison of gross quantities such as critical bank position, excess reactivity, power shapes, etc.)

4.0 METALLURGY

4.1 GENERAL

The Type 3 element was originated for the SM-2, designed by Alco Products, Inc. as a high performance reactor for use by the Army. The SM-2 retains many of the original characteristics, such as core size and configuration, of the original SM-1 reactor but has a longer core life at higher power levels. The SM-2 is a pressurized water facility designed for a core life of 30 MWYR at a power level of 28 tMW. For these design conditions, it was necessary to develop a fuel element which would operate at a maximum surface temperature of 635°F and a maximum centerline fuel temperature of 684°F to an average burnup on the order of 39 percent of the U-235. In addition, the plate was to operate under an average heat flux of 1.65×10^5 Btu/hr-ft² and a maximum heat flux of 4.4×10^5 Btu/hr-ft² with localized areas of the fuel plate having burnups on the order of 82 percent of the U-235.

Previous experience with the SM-1 reactor indicated both the desirability and the feasibility of developing a UO₂-stainless steel dispersion to meet these requirements. Battelle Memorial Institute, subcontractor, to Alco, initiated work to develop a high performance stainless steel fuel element. Several factors were known to be of primary importance:

1. Improvements made in the production of UO₂ fuel since the development of the SM-1 fuel plates made it possible to procure high quality, fully enriched, spherical UO₂ powders from commercial sources.
2. It was desirable to determine whether another type of stainless steel would give better performance than Type 304. The most important factor concerned the incorporation of a known amount of burnable poison in the core.
3. The use of europium-bearing flux suppressors at the end of the active fuel core had been developed at ORNL for use in the SM-1 Core II control rod fuel plates, and this fabrication process could be adapted to the stationary fuel plates.

4.2 UO₂ FUEL

Spherical UO₂ powder was chosen as the reference fuel material, primarily on the basis of two factors:

1. Ceramic-grade UO₂ is known to be relatively free from impurities and to possess excellent resistance to radiation damage. It was shown previously that dispersions superior in all respects to other types of UO₂ available at the time could be made from spherical UO₂. This material

was commercially available at this time in large quantities at prices competitive with the hydrothermal(Geneva oxide) UO_2 used in the SM-1 reactor.

2. The use of spherical UO_2 greatly reduces the fragmentation and stringering of the fuel during the rolling operations utilized in the fabrication of the fuel plates and greatly improves the resistance of the fuel to irradiation damage.

4.3 STAINLESS STEEL CLADDING MATERIAL

Three types of stainless steel were considered as cladding materials: Types 304, 321 and 347. In general, there is little difference among these steels in their nuclear properties and corrosion resistance to pressurized water at 600°F. Type 347 was selected as the cladding material based on the following considerations:

1. Type 304 (Standard or Modified)

This steel is subject to sensitization (carbide precipitation in the grain boundaries). Although modified forms of this steel possessing a low carbon content are relatively free from sensitization, it can occur by diffusion of carbon from the core matrix. Sensitized steel is relatively corrosion resistant to pure 600°F water; but there is always the danger that slight water contamination will lead to intergranular attack and stress corrosion. After the heat treatment required to fabricate a fuel plate by the hot roll bonding method, a large grain size is developed. Coupled with a flatten-annealing operation which causes grain boundary precipitation, this could cause stress corrosion problems.

2. Type 321

This steel is stabilized by additions of titanium. The greatest objection to this steel is that the titanium additions produce a structure characterized by extensive stringers and lapses. The fabricability of this steel with the hot roll cladding process is poor and it did not appear advisable to utilize this material.

3. Type 347

This steel is stabilized with columbium; when properly treated, it is not subjected to sensitization. It has been used so extensively that a comparatively large amount of data concerning its fabricability, corrosion resistance, and irradiation resistance is available. The grain size is inherently smaller than either of the other two.

4.4 CORE MATRIX MATERIAL

A prealloyed Type 347 stainless steel powder was chosen as the matrix material. Blends of elemental chromium, iron and nickel powders were considered but not used because of a nickel-ZrB₂ reaction. At 2300°F, a severe reaction occurred with pure nickel, lending to the formation of a eutectic structure. However, there is no apparent reaction of the ZrB₂ with the prealloyed stainless steel powder.

4.5 BURNABLE POISON MATERIAL

ZrB₂ was selected as the reference burnable poison, followed by NbB₂ as an alternate material. ZrB₂ was found to exhibit suitable low reaction with stainless steel at 2000°F and above, and virtually no boron loss when dispersions were sintered in vacuum at 2150°F. In addition, previous irradiation studies indicated that ZrB₂ has good irradiation resistance and good behavior retention at high B-10 burnups. NbB₂ appeared satisfactory with stainless at 2000°F and exhibited less than 5 percent boron loss when dispersions were sintered at 2000°F. However, little or no information was available on the irradiation behavior of NbB₂.

On the basis of nuclear characteristics, the use of boron as a burnable poison is highly desirable. However, no boron compound is completely stable in stainless steel at the normal fuel plate fabrication temperature of around 2200°F, and boron losses may occur during the fabrication process.⁽⁷⁾ Therefore, in order to select a suitable burnable poison, it was necessary to study the compatibility of various compounds in stainless steel. Although metallographic techniques were used, the selection of the reference compound was based primarily upon boron loss during fabrication as determined by wet chemical analysis.

All known boron compounds were considered and all those with a sufficiently high melting point were selected for investigation, with the exception of the rare-earth borides. These are difficult to obtain and their burnout characteristics would be difficult to ascertain.

In the compatibility studies, 19 borides were used, and after preliminary testing, the most suitable compounds were selected on the basis of metallographic examinations. That is, dispersions containing the boron compound in a stainless steel matrix were sintered in vacuum and under hydrogen atmosphere up to 2300°F and in some cases, roll clad at 2200°F. Compounds which obviously reacted at temperatures on the order of 2000°F were not included in the further studies involving determination of boron loss. Only ZrB₂ gave good results at temperatures above 2000°F and only NbB₂, CrB₂ and BN appeared satisfactory at 2000°F. In all other cases, severe reaction zones on the particles and boron containing phases in the stainless steel matrix could be detected. Additional work on BN and CrB₂ indicated that severe reaction occurred at 2200°F. Therefore, ZrB₂ was selected as the reference burnable poison followed by NbB₂ as an alternate material.

Virtually no boron loss occurred for ZrB_2 dispersion sintered in vacuum at 2150°F and less than 5 percent loss occurred for the NbB_2 dispersions sintered at 2000°F . Analysis of roll clad dispersions indicated that losses could be maintained at less than 10 percent and on the order of 5 percent.

4.6 PREDICTION OF MAXIMUM PERMISSIBLE BURNUP FOR TYPE 3 FUEL PLATES

The maximum permissible irradiation of Type 3 fuel plates was determined by the application of analytical techniques and by the examination of all available irradiation data from many sources on UO_2 stainless steel dispersions.⁽⁸⁾ Two applicable analytical models, one developed by ORNL and the second by Alco, indicated that Type 3 fuel elements will be capable of withstanding 100 percent of U-235. Comparison of analysis with available irradiation data on UO_2 -stainless steel dispersions indicated that successful performance of Type 3 replacement cores in SM-1 and PM-2A can be anticipated.

4.7 FUEL PLATE DEVELOPMENT

4.7.1 Introduction

The 40-mil fuel plate to be used in the Type 3 element was developed by Battelle Memorial Institute under the sponsorship of Alco Products, Inc., as an integral part of the SM-2 Core Materials Development Program.^(7,9,10) Because of time limitations, it was necessary to fabricate irradiation specimens and initiate the irradiation program before much of the development work has been completed. For this reason, the reference-type irradiation specimens were not fabricated strictly according to final fuel element specifications; however, the variables incorporated into the program gave a well rounded irradiation evaluation of the SM-2 Type 3 core materials to be used in SM-1 Core III.

The program consisted of the irradiation of ten capsules containing 62 miniature fuel plates. The first three capsules were non-instrumented capsules and were irradiated in the core of the Materials Testing Reactor (MTR). The last seven capsules contained thermocouples and internal heaters and were inserted in the Experimental Test Reactor (ETR). The three capsules irradiated in the MTR were discharged the last week of March, 1960 and the one capsule irradiated in the ETR was discharged the first week of April, 1960. The other six capsules were inserted between April and September, 1960, and were discharged during October and November, 1961.

4.7.2 Burnup Effects

Irradiation of the initial four of the ten capsules irradiated was successful. The other six capsules have not been examined. The miniature fuel plates in the four capsules achieved a burnup of from 20 to 45 a/o U-235. The burnup for the other six capsules ranged from 65 to 90 a/o U-235.

Twenty-five of twenty-eight specimens irradiated in the capsules experienced maximum density changes of -3.1 percent or less. The general condition of the specimens indicated that the fuel materials should be capable of withstanding higher burnups without serious damage.

During the in-pile exposure, the specimens were subjected to heat fluxes on the order of 10^6 Btu/hr-in² which is considerably greater than the heat fluxes to which the actual SM-1 Core III fuel elements will be subjected. Fission gas was retained within the cladding of all specimens except in those which exhibited failure due to localized overheating. Samples of gas obtained on selected specimens during post-irradiation annealing at 700°F contained no xenon or krypton activity, indicating that cladding integrity was maintained at this operating temperature.

4.7.3 Test Results and Observations

The results of post-irradiation examinations indicated that the Type 3 fuel plate should be capable of withstanding the maximum expected burnup. The following general observations were made:

1. There are no apparent differences in the irradiation behavior of specimens containing Geneva or commercial spherical UO₂.
2. All UO₂ particles showed evidence of sintering. Spherical porosity was observed in the UO₂ particles, the amount and size apparently determined by the amount of burnup and size of UO₂ particle and temperature.
3. In the etched condition, the stainless steel matrix immediately surrounding the UO₂ particles remained unetched. The width of the unetched zones corresponded to the 6 - 10 micron fission recoil range reported for stainless steel.
4. No microcracks were observed in the matrix of the unfailed specimens. The cladding was intact in all specimens examined.
5. Based on metallographic study, the presence of the 22.6 w/o Eu₂O₃ - Type 347 stainless steel flux suppressor does not noticeably affect the behavior of the adjacent fuel core.
6. No adverse effects were noted for specimens incorporating relatively large loadings of burnable poisons in the forms of ZrB₂, NbB₂ and B₄C. No cracking or porosity was observed in the surrounding stainless steel matrix.

The results of visual examination of the miniature fuel plates from the last six capsules should be available by April 1, 1962. Specimens are now at ORNL and are being examined in a hot cell.

4.8 OPERATING EXPERIENCE WITH TYPE 3 ELEMENTS

Two Type 3 fuel elements (SM-2A and SM-2B) were installed into the SM-1 core in July, 1960, at the time that the SM-1 Core I, spiked and rearranged, was loaded. At shutdown in April, 1961, the two elements had been in the reactor for an accumulated core energy release of 1.6 MWYR.

The elements were removed from the core and inspected by periscope in the shield tank. The two elements were reinserted into the core and accumulated an additional 1.4 MWYR, for a total of 3.0 MWYR of service.

The elements were reinspected during the October, 1961, shutdown. In the periscope examination in the shield tank, element SM-2A appeared unchanged from the previous examination. Element SM-2B appeared to have a possible crack progressing along the fuel plate groove on the outside of the side plate. The element was sent to ORNL for hot cell examination. Examination indicated that the suspected cracks were only superficial scratches and the element has been recommended for reinsertion in SM-1 Core II.

4.9 SPECIFICATIONS FOR FABRICATION AND RELATED PROBLEM AREAS

This specification covers the fabrication of both stationary and control rod fuel elements for Core III of the SM-1 and SM-1A reactors. Fuel plates of Type 3 thickness (0.040 in.) have been manufactured at Battelle Memorial Institute. The welded assembly of Type 3 fuel elements was developed at the Alco General Engineering and Test Laboratory.

5.0 SM-1 CORE III THERMAL ANALYSIS

5.1 INTRODUCTION

A preliminary thermal analysis using Type 3 elements in a 37-element PM-2A configuration was performed to determine whether any problems or hazards are present in the operation of the SM-1 Core III. The present SM-1 configuration was reduced to 37-elements by inserting dummy elements in the vacated element positions. The core heat flux at design conditions is 13 percent greater than the SM-1 Core I due to a decrease in effective core heat transfer area resulting from eight less elements and a five rod bank position change. Each of the dummy elements was orificed to receive approximately 75 percent less flow than its respective position in SM-1 Core I. This resulted in modifications to the present core flow distribution and core pressure drop.

The selection of a minimum DNBR for safe operation of the SM-1 Core III is in accordance with the published criteria. A minimum DNBR of 1.5 has been established for steady state and loss of flow transient analyses at design conditions.

The most critical element positions within the core have been analyzed on the basis of an established burnout index. Steady state analyses were performed, using the WAPD code STDY-3,⁽¹³⁾ and transient analyses using the WAPD code ART-02.⁽¹⁴⁾

5.2 METHODS OF ANALYSIS

5.2.1 Steady State Analysis

Core flow distribution, hot channel factors and radial and axial power distributions were established for use in the STDY-3 code. The STDY-3 code utilizes this information to provide necessary data for a complete steady state thermal evaluation of the element for a theoretical hot channel within the most critical SM-1 element. This data includes hot channel mass flow rate, hot channel pressure drop, maximum hot channel surface temperature, maximum hot channel meat centerline temperature, and minimum DNBR. The inception of local or bulk boiling is designated at a particular axial increment up the channel.

5.2.2 Loss of Flow Transient Analysis

A loss of flow transient analysis was performed using the ART-02 code.⁽¹⁴⁾ SM-1 pump failure coastdown data was used. This code subjects the reactor to a variation in flow rate with or without a variation in reactivity due to control rod motion. Reactor kinetic calculations are performed to determine core power as a

function of time. The ART-02 code calculates plate surface temperatures, meat centerline temperatures, heat fluxes, core power and individual channel flow rates as a function of time during flow coastdown for each axial increment up the channel. The minimum DNBR is also calculated and included in the output as a function of time. The important correlations and equations used in this code are presented in the literature.⁽¹⁵⁾

5.3 CORE HYDRAULICS

5.3.1 Core Pressure Drop

The present operating point for primary coolant pump #2 is 4245 gpm at 27.6 ft H₂O total primary system pump head. Element flow is 3295 gpm, core by-pass flow is 315 gpm and lattice flow is 635 gpm. Since SM-1 Core III has a 37-element matrix, Element Positions 12, 16, 21, 27, 61, 72 and 76 are vacant. A dummy element has been inserted in these element positions with an inlet end box orificed to give approximately 10 gpm to each position. An element with this orifice is shown in Dwg.D9-50-1299 at the end of this report. Analyses show that 10 gpm through each of these dummy elements is more than adequate to account for gamma heating effects.

It was assumed that the head versus flow curve for the pump is a straight line in the range under examination. Since the total flow will not change significantly, it was also assumed that the change in total head will be due entirely to the increased pressure drop across the core. Therefore, the portion of the pump curve under consideration can be expressed as a straight line function.

The above analysis was performed for a range of dummy element flows from complete restriction, 0 gpm to 363 gpm. The present total flow through these positions in the SM-1 Core I is 363 gpm. The results of this analysis shown in Table 5.1 indicate for the SM-1 Core III a core Δp of 5.36 ft H₂O and a total pump head of 28.19 ft H₂O with a total flow of 30 gpm flowing through the dummy elements. Core pressure drop for the SM-1 Core I is 4.75 ft H₂O. The SM-1 Core III design represents an increase in core pressure drop of 0.6 ft H₂O and a decrease in loop flow of 63 gpm. A review of this effect on the primary system pump performance has been made. Negligible changes in pump performance are expected with a 0.6 ft H₂O increase in loop pressure drop and a 63 gpm decrease in loop flow.

5.3.2 Core Flow Distribution

A calculation was made to establish the increase in flow through the individual fuel elements caused by orificing the flow through each of the eight dummy elements to 10 gpm. The difference was obtained between the flow through 45-elements and the flow through 37-elements where the flow through the eight dummy elements was restricted to a total flow of 80 gpm. This indicated that the

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individual element flow rates were increased by 6.17 percent. The preceding analysis was based on the fact that the SM-1 core is a tailored core with each element position receiving a prescribed flow. The final SM-1 Core III flow distribution used in the thermal analysis is shown on Fig.5.1. A summary of the pressure drop analysis is shown in Table 5.1.

5.4 POWER DISTRIBUTION

5.4.1 Core Power

In order to place the analysis of the SM-1 Core III on the most conservative basis, both steady state and loss of flow transient analyses were performed at the scram power level. Scram power level is 13.45 Mw for the SM-1. Core average heat flux, peak heat flux, core heat transfer areas for both the SM-1 Core I and SM-1 Core III are shown in Table 5.2.

5.4.2 Axial and Radial Power Distribution

The average and local radial peaking factors listed in Table 5.3 were used to determine the hot channel factors for both the nominal and hot channel. A description of how these peaking factors are used in the thermal analysis is contained in the literature.⁽¹⁵⁾

The average and local radial peaking factors were also used to select the most critical position within the SM-1 Core III. A DNBR index was established⁽¹⁶⁾ using these factors and the individual element flow rates to determine the relative values of DNBR between elements. Tables 5.4 and 5.5 list the DNBR indices for the 37-elements. Asterisks indicate the elements analyzed.

Axial power distribution is shown in Fig. 3.6.

TABLE 5.1
SUMMARY OF PRESSURE DROP ANALYSIS

Δp Core Ft-H ₂ O	Total Flow Thru 8 Corner Positions, GPM	Flow Thru 37 Elements, GPM	Lattice Flow, GPM	By-Pass Flow, GPM	Pump Flow, GPM	Pump Head, Ft-H ₂ O
5.537	0	3165	685	315	4165	28.39
5.355	80	3113	674	315	4182	28.19
5.180	160	3062	663	315	4200	28.03
5.018	240	3013	653	315	4221	27.87
4.75	363	2932	635	315	4245	27.60

TABLE 5.2
CORE POWER FOR SM-1 CORES

	<u>Power, Mw</u>	<u>Core Heat Transfer Area, Ft²</u>	<u>Core Avg Flux, Btu/Hr-ft²</u>	<u>Core Peak Flux, Btu/Hr-Ft²</u>
SM-1 Core I	10.77	562.5	67,650	270,600
SM-1 Core III	10.77	491.7	77,369	299,743

TABLE 5.3
RADIAL POWER GENERATION FACTORS AT 440°F

<u>Element</u>	<u>$Q (\Delta\theta)_O$</u>	<u>$Q (\Delta T)_O$</u>	<u>$\overline{Q (\Delta T)_O}$</u>	<u>$Q (\Delta\theta)_B$</u>	<u>$Q (\Delta T)_B$</u>
13,75	1.856	1.505	.831	1.966	1.594
14,74	1.445	.833	.756	1.530	.882
15,73	1.890	1.536	.838	2.002	1.627
22,66	2.137	1.527	1.038	2.263	1.617
23,65	1.587	1.199	.913	1.691	1.270
24,64-CR	1.613	1.240	.320	1.708	1.313
25,63	1.538	1.183	.946	1.629	1.253
26,62	2.217	1.629	1.059	2.348	1.725
31,57	1.957	1.454	.916	2.072	1.540
32,56	1.627	1.362	.956	1.723	1.442
33,55-CR	1.784	1.386	.369	1.889	1.468
34,54	1.801	1.409	1.105	1.907	1.492
35,53	1.685	1.267	1.108	1.784	1.342
36,52	1.492	1.167	.965	1.580	1.236
37*	2.186	1.562	.957	2.315	1.654
51*	2.150	1.522	.934	2.277	1.612
41	1.442	1.304	.824	1.527	1.527
42-CR	1.523	1.254	.323	1.613	1.613
43	1.671	1.326	1.124	1.770	1.770
44-CR	1.719	1.430	.427	1.820	1.820
45	1.675	1.337	1.127	1.774	1.774
46-CR	1.537	1.261	.325	1.628	1.628
47	1.450	1.317	.828	1.536	1.536

* Denotes special elements

TABLE 5.4
SUMMARY OF THE I_{DNBR} INDICES FOR SM-1 CORE III, CORE POSITIONS 12-43

Core Position	$Q\Delta T$	G, gpm	$Q\Delta\theta$	I_{DNBR}
12	Dummy	10	Dummy	
13	1.594	59.53	1.966	2.844
14	.882	63.03	1.530	1.794
*15	1.627	60.97	2.002	2.893
16	Dummy	10.00	Dummy	
21	Dummy	10.00	Dummy	
22	1.617	71.20	2.263	3.107
23	1.270	94.94	1.691	2.068
24	1.313	99.81	1.708	2.084
25	1.253	93.40	1.629	1.995
26	1.725	70.62	2.348	3.304
27	Dummy	10.00	Dummy	
*31	1.540	58.51	2.072	2.980
32	1.442	92.79	1.723	2.168
*33	1.468	94.37	1.889	2.380
*34	1.492	106.75	1.907	2.351
35	1.342	106.43	1.784	2.159
36	1.236	92.37	1.530	1.932
*37	1.654	60.15	2.315	3.375
41	1.527	60.97	1.527	2.164
42	1.613	100.18	1.613	2.045
43	1.770	112.29	1.770	2.236

TABLE 5.5
SUMMARY OF THE I_{DNBR} INDICES FOR SM-1 CORE III, CORE POSITIONS 44-76

Core Position	$Q\Delta T$	G, gpm	$Q\Delta\theta$	I_{DNBR}
*44	1.820	99.81	1.820	2.371
45	1.774	111.26	1.774	2.244
46	1.628	97.51	1.628	2.081
47	1.536	66.54	1.536	2.127
*51	1.612	62.00	2.277	3.263
52	1.236	91.35	1.580	1.936
53	1.342	103.05	1.784	2.171
54	1.492	112.29	1.907	2.330
55	1.468	94.16	1.889	2.380
56	1.442	93.61	1.723	2.166
57	1.540	60.76	2.072	2.946
61	Dummy	10.00	Dummy	
*62	1.725	71.85	2.348	3.287
63	1.253	98.13	1.629	2.052
64	1.313	100.18	1.708	2.080
65	1.270	91.97	1.691	2.080
*66	1.617	71.93	2.263	3.112
67	Dummy	10.00	Dummy	
72	Dummy	10.00	Dummy	
73	1.627	62.00	2.002	2.877
74	.882	62.41	1.530	1.890
*75	1.594	53.16	1.966	2.949
76	Dummy	10.00	Dummy	

		13	14	15		
		10.0	59.7	63.3	61.2	10.0
10.0	22	23	100.2	25	26	10.0
		71.3	95.2	93.7	70.9	
31	32	94.7	34	35	36	37
		93.1	107.1	106.8	92.7	60.4
41		43	100.2	45		47
	100.6	113.7	100.2	111.6	97.8	66.8
51	52	53	54		56	57
		91.7	103.4	112.7	94.5	93.9
						70.0
10.0	66	63	100.5	65	66	10.0
		72.1	98.5	92.3	72.2	
		73	74	75		
		10.0	62.2	62.6	53.3	10.0

Note: Values shown in element positions are in GPM
 3113.9 - GPM total internal fuel element flow plus
 674 - GPM lattice flow
 80 - GPM total dummy element flow

Figure 5.1. SM-1 Core III Flow Distribution

5.5 HOT CHANNEL FACTORS

The mechanical hot channel factors used in the steady state and transient analysis include average and local deviations for meat length, uranium content and clad thickness. The preceding factors were calculated by methods described in the literature.⁽¹⁵⁾

Plate spacing appears in the hot and nominal description as an average and local dimension, not as an individual factor. The average and local dimension for the nominal channel is 0.123 inches. A ripple growth factor of 1.3 was applied to the stationary element local dimension and 1.10 to the control rod to obtain maximum local dimensions of 0.136 and 0.133 in. respectively.

Channel-to-channel maldistribution factors were obtained from single element flow testing of Type 3 elements.⁽¹⁷⁾

To allow for additional conservatism in the analysis, a nuclear uncertainty factor, a core power generation factor (to account for the fraction of the heat liberated in the fuel plates) and instrument tolerances were included.⁽¹⁵⁾

5.6 PERFORMANCE OF SM-1 CORE III

5.6.1 Steady State Analysis

A steady state analysis were performed on the 10 most critical elements within SM-1 Core III. The selection of these elements was based on the burnout indexes presented in Section 5.4.2. These elements appear in Table 5.6.

At design conditions, reactor power was 10.77 Mw with a core heat flux of 77,369 Btu/hr-ft² and inlet temperature of 431.7°F. At scram power conditions, reactor power was 13.45 Mw with a core heat flux of 93,388 Btu/hr-ft² and inlet temperature of 429°F. Hot channel plate surface temperature, hot channel meat centerline temperature, hot channel bulk water temperature, departure from nucleate boiling ratio, local nucleate boiling, bulk boiling and percent quality were calculated for one inch increments up the hot channel. The results of this analysis are shown in Table 5.6.

The results of this analysis indicate minimum DNBR's at design conditions are above 1.5, the currently specified minimum criteria for steady state and loss of flow transient analysis.⁽¹⁵⁾

The high burnup elements located in positions 37 and 51 indicate both local nucleate and bulk boiling at scram power in their hot channels. Peripheral elements 75, 15 and 62 also indicate both local nucleate boiling and bulk boiling at scram power. The average channels of these elements which were analyzed at design and scram power did not show bulk boiling. At design power, 10.77 Mw, local nucleate boiling occurred in all cases analyzed except the control rod positions 33 and 44 and stationary element position 34. Element positions 33, 44, and 34 did

TABLE 5.6
SUMMARY OF THE RESULTS OF THE STEADY STATE THERMAL
ANALYSIS OF SM-1 CORE III

Core Position	Core Power, Mw	Nominal Channel-G lb/hr-ft ²	Hot Channel Bulk Outlet Temp, °F	Hot Channel Max Plate Surface, °F	Minimum DNBR	Hot Channel Local Boiling	Hot Channel Bulk Boiling	% Quality
37	10.77	6.246 x 10 ⁵	544	576.4	4.365	J=L	0	0
	13.45		565	577.0	3.530	J=3	J=19	2.1
51	10.77	6.461 x 10 ⁵	529	576.3	4.493	J=5	J=0	0
	13.45		555	577.0	3.652	J=3	J=22	.4
62	10.77	7.460 x 10 ⁵	515	576.53	4.406	J=6	J=0	0
	13.45		565	576.93	3.530	J=3	J=19	2.1
66	10.77	7.495 x 10 ⁵	504	576.34	4.626	J=0	J=0	0
	13.45		522	576.88	3.788	J=5	J=0	0
31	10.77	6.097 x 10 ⁵	530	576.22	4.935	J=5	J=0	0
	13.45		557	576.75	4.010	J=3	J=0	0
75	10.77	5.540 x 10 ⁵	542	576.30	5.153	J=22	J=0	0
	13.45		565	576.83	4.174	J=3	J=20	1.4
15	10.77	6.354 x 10 ⁵	541	576.30	5.068	J=22	J=0	0
	13.45		565	576.90	4.103	J=3	J=20	1.1
33	10.77	10.80 x 10 ⁵	465.2	576.1	5.001	J=0	J=0	0
	13.45		468	576.6	4.076	J=0	J=0	0
44	10.77	11.42 x 10 ⁵	471	576.7	5.090	J=0	J=0	0
	13.45		475	577.2	4.197	J=19	J=0	0
34	10.77	11.15 x 10 ⁶	474	564.7	5.619	J=0	J=0	0
	13.45		479	576.6	4.705	J=0	J=0	0

* Axial increment measured from the bottom of the core at which local boiling commences

not show either bulk or local nucleate boiling due to higher element flow rates than those of the peripheral elements and better internal channel to channel flow distribution.

Figures 5.2 and 5.3 show for element position 37 at design conditions and scram power, the variation of meat centerline temperature, plate surface temperature and bulk water temperature with position for the hot channel. The inception of local nucleate boiling and bulk boiling is noted for each case.

In comparison with SM-1 Core I, SM-1 Core III contains fewer active fuel elements. This serves to decrease the core heat transfer area and results in an increase in core heat flux for the same power output. The results of the thermal analysis performed on element position 44 for both cores are listed in Table 5.7.

TABLE 5.7
COMPARISON OF THERMAL PERFORMANCE OF ELEMENT POSITION 44
IN SM-1 CORES I AND III

<u>Core</u>	<u>Power Level, Mw</u>	<u>Flow, GPM</u>	<u>$Q\Delta T$</u>	<u>$Q\Delta\theta$</u>	<u>Core Heat Flux, Btu/Hr-Ft²</u>	<u>Hot Channel Max Plate Surface Temp, °F</u>	<u>Hot Channel Bulk Outlet Temp, °F</u>	<u>Min DNBR</u>
SM-1 Core II	10.77	94	1.48	1.58	67,650	530.31	434.0	7.61
SM-1 Core III	10.77	100	1.82	1.82	77,369	576.70	471.0	5.09

The results of the steady steady state analysis indicate that the SM-1 Core III will operate safely during design conditions and scram power conditions. This is based on DNBR's being above the minimum criteria of 1.5. The existence of local nucleate boiling at design conditions and bulk boiling at scram conditions is noted.

5.6.2 Transient Analysis

The minimum allowable DNBR in a reactor loss of flow transient has been set at 1.5. The loss of flow transient analysis was performed on the most critical element in SM-1 Core III, element position 37. To insure the conservatism of the transient analysis the power level used was 13.45 Mw. If this element proves safe to operate at this level it can be assumed that the entire SM-1 Core III will be safe thermally during a loss of flow transient.

The analysis on this element assumed a "no scram" condition with the most adverse pump coastdown. A plot of flow coastdown as a function of time is given on Fig.5.4. This pump coastdown is considerably more severe than that measured at SM-1 during TP-600. (18)

Figure 5.2. Hot Channel Temperatures Vs Length Along Channel for Element Position 37 at Reactor Power 10.77 Mw

Figure 5.3. Hot Channel Temperatures Vs Length Along Channel for Element Position 37 at Reactor Power 13.45 Mw

Figure 5.4. Flow Coastdown Investigation in Loss of Flow Analysis

Figure 5.5. Minimum Burnout Ratio (DNBR) for Hot Channel Vs Time After Loss of Flow for Element Position-37

The minimum DNBR results for a coastdown of 5 sec are shown on Fig. 5.5. The effect of variation in flow coastdown on thermal conditions is shown in Table 5.8.

The results of this analysis indicate that during the critical period immediately following a loss of flow accident the minimum DNBR is 3.74. The reactor will scram because of low flow in approximately .050 sec.

TABLE 5.8
SUMMARY OF RESULTS OF LOSS OF FLOW ANALYSIS
ELEMENT POSITION 37

<u>Time, Sec</u>	<u>Hot Channel Max Bulk Temp, °F</u>	<u>Hot Channel Max Surface Temp, °F</u>	<u>Minimum DNBR</u>	<u>Inception of Bulk Boiling J = Axial Increment</u>	<u>Hot Spot, % Quality</u>
0	564.2	577.5	3.74	J = 21	1.0
.025	564.2	577.5	3.74	J = 21	1.0
.5	562.3	577.0	3.74	J = 16	2.8
1	562.2	576.1	5.90	J = 17	4.1
2	558.7	575.0	9.02	0	0
3	561.7	574.3	9.28	0	0
4	562.5	573.8	11.80	J = 18	1.5
5	561.5	573.5	13.1	J = 19	1.8

5.7 CONCLUSIONS AND RECOMMENDATIONS

1. The preliminary thermal analysis indicates the SM-1 Core III will operate safely for steady state and loss of flow transient conditions at design power 10.77 Mw and scram power 13.45 Mw.
2. The minimum DNBR for steady state conditions at 13.45 Mw is 3.53. The minimum DNBR for the loss of flow transient condition at 13.45 Mw is 3.74. Both of these conditions indicate DNBR's well above the minimum requirement of 1.5 for both the steady state and transient analyses.
3. Steady state analysis indicated local nucleate boiling in the hot channels of element positions 37, 51, 62, 31, 75, 15, at design power 10.77 Mw. Elements 37, 51, 31, 75 and 15 are located on the periphery of the core. A limited amount of bulk boiling was found in the exit end of the hot channels within elements 37, 51, 62, 75 and 15 at scram conditions 13.45 Mw. Bulk boiling was not found in any of the nominal channels analysed at steady state.

4. During the loss of flow transient analysis bulk boiling was evident in element position 37 during 0.25 to 1.7 sec after the start of pump coast-down. Bulk boiling did not appear in the hot channel until 1.6 sec later and continued for the remainder of the five sec transient. Minimum DNBR occurred at the start of the flow coastdown.
5. Good agreement was found between the steady state minimum DNBR of 3.53 and the transient DNBR of 3.74 at 0 seconds. This small difference results from input format differences in the two codes and small variations in the application of the DNB correlations.

As a result of this analysis the following recommendations are made:

1. Perform thermal analysis of the hottest lattice passage between two stationary elements.
2. Re-analyze the transient loss of flow accident transient analyses for loss of flow accident should be updated with the use of the ART-04 code. This updating is due to the fact that the ART-04 code utilizes an advanced slip flow model instead of a fog flow model used in the ART-02 code.
3. Due to the existence of a considerable amount of local nucleate boiling and bulk boiling in the peripheral elements, a density coefficient should be incorporated in future thermal analyses to account for reactivity effects caused by boiling.
4. Analysis by metallurgy and radiochemistry to determine effects of nucleate boiling on cladding material.
5. Removal of high elements or relocation is recommended if the core analysis shows a reduction in radial peaking factors as a result of this. A reduction in power peaks will reduce or eliminate bulk boiling in the hot channels.

6.0 OPERATIONAL CONSIDERATIONS

The installation of SM-1 Core III containing Type 3 elements in the SM-1 necessitate some minor changes in established operating procedures at the SM-1. An evaluation of the effect of these changes on the safety of the system is given below. No significant changes in operation are anticipated for the Type 3 core in PM-2A.

6.1 SHUTDOWN PROVISIONS

The SM-1 Core III loading and operation will simulate conditions in the PM-2A by actively using only five of the seven control rods provided in the SM-1. Rods A and B will not be actively used with Core III but will be fully withdrawn. The control rod fuel elements for rods A and B will remain in place and simulate PM-2A stationary fuel elements in these positions.

Rods A and B will remain in the fully withdrawn position during all operations with SM-1 Core III. Procedures will be modified to insure that control rods A and B are fully withdrawn before the five active control rods are withdrawn. Rods A and B will remain in the scram circuits and will be available as an additional means of shutdown.

6.2 ROD PROGRAMMING

At the SM-1, standard operating procedures allow pulling rods in a seven rod bank during an attempt to recover power following a scram. These procedures will be modified to prohibit seven rod bank operation. Administrative control (as dictated by the revised operating procedures) will require that rods A and B be completely withdrawn before the five active rods are allowed to be withdrawn.* It will be allowable for rods A and B to be withdrawn simultaneously, and following this action, the remaining rods will be permitted to be withdrawn to within one inch of their former critical bank position. Only single rod movement will be allowed after this point is reached, and subsequent startup will proceed according to procedures formerly used.

* This will cause a delay of seven min while rods A and B are raised and will probably make it impossible to startup after a scram on the log N channel.

6.3 NUCLEAR INSTRUMENTATION RESPONSE

The SM-1 Core III and the Type 3 core in the PM-2A will have a different neutron flux density for a given power level as compared to SM-1 Core I and PM-2A Core I. These differences are difficult to quantify at this time, but it is thought that startup will be faster for a given amount of control rod travel. For this reason, the operators will need to become accustomed to a slightly different action of the instrumentation during startup. As criticality is approached, control rods should be pulled at a slower rate than normal until the operators become familiar with the instrumentation response with the Type 3 element loading.

7.0 SHIELDING

7.1 DOSE RATES DURING OPERATION AND SHUTDOWN

Radiation dose rates from SM-1 Core I in plant areas where access is required are presented in the Hazards Summary Report for SM-1, APAE-2, Rev. 1.⁽¹⁹⁾ An SM-1 core with Type 3 elements (Core III) will operate at the same power level as SM-1 Core I (10.8 Mw). Differences between the two cores which affect the operating and shutdown dose rates are small and tend to cancel each other. For example, Core III has a higher source strength than Core I but has about one more inch of water shielding because of the smaller volume of Core III relative to Core I. The total accumulated fission product inventory at the end of SM-1 Core III life will be about 0.5 percent greater than that of Core I. This difference results from the increased lifetime of Core III (24 MWYR predicted), compared with a measured 16.4 MWYR for Core I. Therefore, the dose rates presented in APAE-2, Rev. 1,⁽¹⁹⁾ for SM-1 Core I will also be applicable to Core III.

Operation of Type 3 elements in PM-2A will likewise result in negligible changes in the PM-2A plant radiation levels during either operation or shutdown. The Type 3 core in PM-2A, with an anticipated 24 MWYR life, will contain less than a 1 percent greater total fission product accumulation than Core I will contain at the end of its life which is predicted to be 10.7 MWYR. Therefore, the dose rates presented for PM-2A Core I⁽²⁰⁾ will also be applicable to this core.

7.2 DOSE RATES DURING REFUELING

SM-1 Core III contains eight less fuel elements than Core I. Therefore, the fission product inventory per Core III fuel element will be about 20 percent greater than that of a Core I element having an equivalent history; consequently, the dose rate from a Core III fuel element will be no more than 20 percent higher than that from a Core I element. The values given in Table V-8 of the Hazards Report⁽¹⁹⁾ will accordingly be 20 percent lower than if Type 3 elements were employed. This increase should not present an operating problem or a hazard to personnel performing the SM-1 refueling operation.

The Type 3 core in PM-2A will contain the same number of fuel elements as Core I. The fission product inventory per Type 3 fuel element will be less than 1 percent higher than the inventory in a Type 1 fuel element. This will produce an insignificant increase in the PM-2A refueling dose rates.

8.0 SPECIAL CONSIDERATIONS

8.1 POST-IRRADIATION EXAMINATION OF DESIGNATED FUEL ELEMENTS

Several elements will be earmarked during the fabrication stage for post-irradiation examination. These elements will be designated by number and will receive special attention during production inspection to obtain complete pre-irradiation dimensions and other data. They will then be loaded in specified high flux core positions.

8.2 EFFECT OF TYPE 3 ELEMENTS ON REACTOR VESSEL RADIATION DAMAGE

The nvt/MWYR on the pressure vessel of the SM-1 plant utilizing a Type 3 core of 37-elements rather than a Type 1 core of 45-elements has been investigated using a cylindrical model. With this model, the 37-element core has an effective radius of 10.8 in., while the 45-element core has an effective radius of 11.1 in. thereby giving the smaller core an extra 1.02 in. of water between the core and pressure vessel. Although the smaller core will have a higher leakage of fast flux greater than 1 Mev, the added attenuation of 1.02 in. of water will more than compensate for this increase. Estimates show that a decrease of approximately 30 percent in total nvt/MWYR may be realized. It must be emphasized, however, that this figure was obtained with the cylindrical model, whereas the actual physical cores are rectangular configurations (see Fig. 3.1 and 3.3). In the rectangular configuration, the distance from the sides of the core to the pressure vessel is the same for both the 37-element and the 45-element cores. Thus, there is no extra water attenuation and since the smaller core has a higher power density, the nvt/MWYR might actually increase at the regions of the pressure vessel near the sides of the core. The effect is different at the corners of the cores, where there is a difference in the distance to the pressure vessel between the 37- and 45-element cores.

A two dimensional (x,y) multi-group analysis of this problem is planned to obtain more accurate values of nvt on the pressure vessel of the SM-1 Core III with the Type 3 elements.

The PM-2A core with Type 3 elements should expose the pressure vessel to the same nvt/MWYR, plus or minus 5 percent, as the PM-2A core with Type 1 elements because the geometries of the two configurations are the same and composition changes of this type (Type 1 to Type 3 elements) have little effect on nvt. (21, 22)

8.3 EFFECT OF TYPE 3 ELEMENTS ON SPENT FUEL CRITICALITY

The loading of Type 3 elements in the spent fuel pit for the SM-1 should be examined to see if there is a criticality problem. The configuration and the

cadmium lining in the cylindrical fuel element holders should prevent criticality, but an inverse multiplication experiment is recommended if the spent fuel pit is to be used for storing fresh elements. A similar analysis should be made of the effect of loading Type 3 elements in the PM-2A fuel pit.

8.4 EFFECT OF TYPE 3 ELEMENTS ON REPROCESSING

The changing of the clad and matrix material from Type 304 to Type 347 stainless steel will have no effect on either the dissolution of the fuel plates or the cost of reprocessing the spent fuel elements. The fuel elements are dissolved in acid and the less than 1 percent columbium in the Type 347 stainless steel does not appreciably increase the corrosion resistance to acids.

8.5 SHIPMENT CONSIDERATION

Shipping containers for fresh fuel elements and shipping casks for spent fuel, similar to those used for Type 1 elements in both the SM-1 Core I and the PM-2A Core II, may now be used for Type 3 elements in SM-1 Core III and for Type 3 elements in the Type 3 core for PM-2A, on the basis of current government regulations.

New regulations covering the shipment of radioactive materials are presently under consideration and may be in force at the time of shipment of a fresh core and a spent core. Prior to placing an order for fabrication of containers for fresh Type 3 elements, a review of the container design would have to be made on the basis of the additional tests and requirements on containers as set forth in reference (23). Prior to placing an order for fabrication of shipping casks for a spent core of Type 3 elements, the proposed regulation 10 CFR 72, (24) revisions to AEC manuals on shipments of spent cores, (25) and "ICPP Fuel Receiving Facility and Cask Design Criteria" (26) would also have to be considered.

9.0 REACTOR SAFETY

9.1 EFFECT ON POSTULATED NUCLEAR EXCURSIONS

The hazards analyses for both the SM-1 and the PM-2A plants^(19,20) presented calculations for the maximum energy releases which could be expected for various nuclear excursions. For the SM-1 plant, the cases considered were:

1. Continuous rod withdrawal.
2. Inadvertent insertion of a fresh fuel element in a central water hole of a just critical core.
3. Full rod ejection following rupture of the outlet pipe at the exit from the shield tank.

The primary factors contributing to the severity of the postulated excursions are the rod bank worths, the rod withdrawal rates, core mean neutron lifetime, and the total available excess reactivities available in the cores under consideration. Other plant parameters play an essential role, but these would not be changed by the substitution of Type 3 elements in the cores for these plants.

Table 9.1 below gives the nuclear factors considered to be of primary importance and lists them before and after the substitution of Type 3 elements.

TABLE 9.1
REACTIVITY COMPARISON, SM-1 AND PM-2A,
WITH SUBSTITUTION OF TYPE 3 ELEMENTS

<u>Core Nuclear Data</u>	<u>SM-1*(Core I)</u>	<u>SM-1/Type 3 (Core III)</u>	<u>PM-2A (Core I)</u>	<u>PM-2A/Type 3</u>
Rod bank worth, k	25%	15.4%	21.7%	22.1%
Reactivity insertion rate	1.3×10^{-3} $\Delta k/\text{sec}$	7.9×10^{-4} $\Delta k/\text{sec}$	1.18×10^{-3} $\Delta k/\text{sec}$	1.2×10^{-3} $\Delta k/\text{sec}$
Core mean neutron lifetime	2×10^{-5} sec	1×10^{-5}	2.6×10^{-5} sec	1×10^{-5}
Total excess k, cold	15.4%	10.9%	14.3%	12.1%

* SM-1 values are given for seven rod bank, as used in the hazards analysis. All other values are for a five rod bank.

As can be observed from this table, the substitution of Type 3 elements will not radically alter the critical nuclear parameters. The substitution of Type 3 elements shortens the mean neutron lifetime, but this is compensated for by the lower excess reactivity of the cores containing the Type 3 elements. It is concluded, therefore, that the Type 3 element cores have nearly the same self shut-down characteristics as Type 1 element cores.

9.2 EFFECT ON POSTULATED MAXIMUM CREDIBLE ACCIDENT

The maximum credible accident for the SM-1 and the PM-2A plants is defined as a failure occurring simultaneously in their primary and secondary systems resulting in a release of all of the stored energy in these systems to the vapor container volume. The loss of coolant exposes their reactor cores, causing fuel melting and release of the fission products which have accumulated during their entire core operating lifetimes. The energy stored in the primary and secondary systems in these two plants is greater by a factor of 100 than the values calculated for the maximum energy release as a result of nuclear excursions. Therefore, it is concluded that substitution of Type 3 elements in these plants should have no significant effect on the total energy release of the maximum credible accident.

9.3 EFFECT ON FISSION PRODUCT RELEASE

In the SM-1 plant, the total fission product inventory for Core III will be the same as that for Core I for identical operating histories; however, the total accumulated fission product inventory at the end of Core III life is expected to be slightly greater (less than 0.5 percent) than that of Core I. This difference results from the increased lifetime of Core III (24 MWYR predicted) compared to the 16.4 MWYR measured for Core I. The slightly greater fission product inventory in Core III would, therefore, result in a slightly greater, but not significantly greater fission release in the event of the maximum credible accident in the SM-1.

For the PM-2A plant, the same conclusion applies. The total accumulated fission product inventory at the end of life of a Type 3 core in PM-2A will be less than 1 percent greater than that of Core I. This difference results from the increased life of the Type 3 core in PM-2A (24 MWYR predicted) compared to a predicted 10.7 MWYR for Core I. This will not result in a significantly greater fission product release in the event of the maximum credible accident in the Type 3 core in PM-2A.

10.0 CONCLUSIONS AND RECOMMENDATIONS

1. Heavier loaded cores demand a more exact determination of contained burnable poison in order to meet startup conditions and the one stuck rod criterion. This determination requires additional analysis which is being undertaken.
2. The Type 3 element is metallurgically sound.
3. The Type 3 core elements will operate safely within specified thermal safety requirements (DNBR greater than 1.5). However, due to higher power densities, the hottest lattice passages need to be analyzed to guarantee safe performance. This analysis is being accomplished.
4. The use of a 37-element core in the SM-1 will not significantly alter the operation of the SM-1.
5. Dose rates through existing shielding are not expected to change.
6. The NDT considerations are not significantly altered.
7. The spent fuel pits at the SM-1 and PM-2A should be re-examined to guarantee that they can hold a fresh core of Type 3 elements.
8. Shipment considerations do not present a problem.
9. The reactor safety considerations as presented in the hazards summary reports have not significantly changed.

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