APAE No.-84
Copy No. 20
AEC Research and
Development Report
UC-81, Reactors-Power
(Special Distribution)

**HAZARDS REPORT FOR THE SM-1 CORE II
WITH SPECIAL COMPONENTS**

By:
J. Coombe I. Segalman
D. Lee R. Robertson

Compiled By: J. Coombe

Approved By: H. L. Hoover, Project Engineer

Issued: March 30, 1961

Contract No. AT-(30-1)-2639, with
the United States Atomic Energy Commission,
New York Operations Office

ALCO PRODUCTS, INC.
Nuclear Power Engineering Department
Post Office Box 414
Schenectady 1, N. Y.

ABSTRACT

This report describes the changes incurred in the SM-1 by the insertion of the SM-1 Core II and special components. The special components consist of impact specimens, a boron gradient rod, SM-2 elements, a PM-1-M element, and high burnup SM-1 Core I elements. The change in hazards, due to operation of SM-1 with Core II and the special components, is evaluated. The analysis indicates there is no change in hazards.

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	vii
1.0 INTRODUCTION	1
2.0 CORE II COMPONENTS	3
2.1 Description	3
3.0 INVENTORY OF SPECIAL COMPONENTS	11
3.1 SM-2 Elements	11
3.1.1 SM-2A Stationary Fuel Element	11
3.1.2 SM-2B Stationary Fuel Element	11
3.1.3 Testing Procedures, SM-2 Element	15
3.1.4 Summary	16
3.2 Two High Burnup Core I Stationary Fuel Elements	16
3.2.1 Irradiation Blistering	16
3.2.2 Fuel Plate Rippling	16
3.2.3 Loss of Comb Due to Braze Metal Corrosion	17
3.2.4 Loosening of Fuel Plate Due to Braze Metal Corrosion	17
3.2.5 Clad Cracks and Waterlogging	17
3.2.6 Activity Increase from Fission Product Escape Through Clad Cracks	17
3.2.7 Summary	17
3.3 SM-1 Core II Materials Test Package	18
3.3.1 Description	18
3.3.2 Impact Specimens	18
3.3.3 Flux Monitors	18
3.3.4 Temperature Indicators	27
3.3.5 Fabrication	27
3.3.6 Justification for Insertion in Core II	28
3.4 PM-1M Fuel Element	28
3.5 Boron-Gradient Absorber	28
3.5.1 Summary	29

TABLE OF CONTENTS (CONT'D)

	<u>Page</u>
4.0 NUCLEAR CHARACTERISTICS-----	31
4.1 Introduction -----	31
4.2 Core Reactivity -----	31
4.3 Control Rod Worth -----	31
4.4 Local Power Peaking -----	34
4.5 Conclusions -----	34
5.0 THERMAL CHARACTERISTICS -----	35
5.1 Minimum Burnout Ratios -----	35
5.2 Input Parameters -----	35
5.3 Steady State Analysis -----	37
5.4 Conclusions -----	41
5.5 Transient Analysis -----	41
5.5.1 Discussion -----	41
5.5.2 Conclusions -----	41
6.0 MAXIMUM TEMPERATURE IN MATERIALS TEST PACKAGES --	47
6.1 Temperature Determination -----	47
6.2 Conclusion -----	47
7.0 DISCUSSION AND CONCLUSIONS -----	49
BIBLIOGRAPHY -----	51

LIST OF ILLUSTRATIONS

<u>Figure</u>	<u>Title</u>	<u>Page</u>
2.1	Control Rod Fuel Element, Core II	5
2.2	Control Rod Assembly, Core II	7
2.3	Control Rod Absorber Section, Core II	9
3.1	SM-1 Core II, with Special Test Elements	13
3.2	Dwg. R9-50-1112 - SM-2 Stationary Fuel Element	19
3.3	Dwg. AEL-644 - Fuel Element Stationary - Dummy	21
3.4	Capsules for Irradiation Experiment	23
3.5	Radiation Exposure Container	25
5.1	Element Nos. 65 and 52, Hot Channel Plate Surface Temperature vs. Length Along Plate, 13.45 MW	39
5.2	Element Nos. 65 and 52, Hot Channel Plate Surface Temperature vs. Length Along Plate, 10.77 MW	40
5.3	Burnout Ratio vs. Time for Element No. 65	48

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
3.1	Boron Loading In Experimental Absorber BG-1	30
4.1	Material Composition of Core II Components	32
4.2	Stuck Rod Measurements in SM-1 Core II with Special Component Mockups	33
5.1	Summary of Hot Channel Factors	36
5.2	Summary of Basic Input Parameters for the Elements Investigated	37
5.3	Summary of Results of Steady State Analysis	38
5.4	Transient Analysis of Stationary Element 65 for Initial Core Powers of 10.77 MW and 13.45 MW	45

1.0 INTRODUCTION

The Army Package Power Reactor (now known as the SM-1) was designed and constructed by Alco Products, Inc., under Contract No. AT(11-1)-318 with the Army Reactors Branch of the AEC. The SM-1 is a 10 tMW pressurized water reactor of a prototype nature designed to meet the requirements and site conditions of a remote military base. It is located at Fort Belvoir, Virginia, within the bounds of the Engineer Research and Development Laboratory.

The original SM-1 core, called Core I, is now being replaced by modified Core II. An evaluation of this Core II is given in APAE No. 2. ⁽¹⁾ This core was made to essentially identical specifications as the original core, and the zero power experiment ⁽²⁾ demonstrated that the nuclear characteristics were also essentially identical.

A group of special components is scheduled to be included with the installation of Core II. An initial survey of these components ⁽³⁾ and a complete core analysis have been made. ⁽⁴⁾ This report will examine the special components with reference to their hazard potential in Core II.

2.0 CORE II COMPONENTS

2.1 DESCRIPTION

Basic differences between SM-1 Core I elements and Core II elements are:

Fixed Elements

1. Slight increase of B-10 loading (as B_4C) to 0.4163 gm/element from 0.3605 gm/element.
2. Cladding is 304L ss with lower cobalt content.

Control Rod Elements (Figs. 2.1 and 2.2)

1. Cladding is 304L ss with lower cobalt content.
2. Integral flux suppressor is Eu_2O_3 dispersed in stainless steel at the top of the fuel section of the control rod, replacing the Haynes Alloy combs on the SM-1 Core I fuel elements.

Absorber Section (Fig. 2.3)

1. Use of Eu_2O_3 stainless steel instead of boron-iron alloy enriched in B-10 as absorber material.

A comprehensive statement of the differences is given in Reference (1).

The lower cobalt content, present as an impurity, does not affect the metallurgical properties of the cladding. The use of low impurity core materials is a consequence of the buildup of long-lived radioactive nuclides on primary system surfaces and the resultant maintenance problem.

The performance of SM-1 Core II control rod elements with the Eu_2O_3 flux suppressors has been successfully demonstrated by the operation of one element of this type in SM-1 Core I from 10.5 through 18 MWYR. In addition, irradiation tests of specimens containing 20, 30, and 40 w/o Eu_2O_3 in stainless steel have been conducted in the ETR. (5) This program is well under way toward completion and no evidence of irradiation failure has appeared. (6)

Figure 2.1 Control Rod Fuel Element, Core II

Figure 2.3 Control Rod Absorber Section, Core II

3.0 INVENTORY OF SPECIAL COMPONENTS

Several special components are planned for insertion in the SM-1 Core II, and will be positioned as shown in Fig. 3.1. All of these components contribute to the development of improved replacement cores for the SM-1, SM-1A, and PM-2A, and will also contribute to the progress of the entire Army Nuclear Power Program. A description and analysis of the reasons for the insertion and the hazard potential for each component follows.

3.1 SM-2 ELEMENTS

3.1.1 SM-2A Stationary Fuel Element

SM-2A is a stationary SM-2 fuel element containing 49.485 gm U-235 per plate and 0.09478 gm B-10 per plate. At the end of Core I life, the element was used for spiking the rearranged Core I. During operation in the rearranged and spiked core, the element received an estimated average burnup of 2 percent U-235 and 10 percent B-10.

3.1.2 SM-2B Stationary Fuel Element

The SM-2B element embodies nine different concepts in fuel plate design. The total loading in the element is 886.56 gm U-235 and 1.588 gm B-10. Two of each of the following plates have been included in the element:

1. A reference SM-2 plate. This plate contains 49.45 gm U-235 and 0.103 gm B-10 as ZrB_2 .
2. 51.79 gm U-235 are contained in a UN composition coated with Cb. The plate contains 0.107 gm B-10 as ZrB_2 .
3. A basic SM-1 plate has been scaled up to SM-2 thickness (0.040 in.); 43.24 gm U-235 and 0.043 gm B-10 as B_4C are contained in this plate.
4. 51.61 gm U-235 in the form of UN has been used to fuel two plates. These plates contain no boron.
5. Two plates with 49.45 gm U-235 in a spherical UO_2 compound have been included, in which 0.04 gms of B-10 have been alloyed in the stainless steel frame.
6. A "pressed only" core in otherwise reference SM-2 plates. Fuel loading is 49.28 gm U-235 and 0.102 gm B-10.

	12 IMPACT SAMPLES	13	14	15	16	
21	22	23	24	25	26 PM-IM	27
31	32	33	34	35 HIGH BURN UP CORE I ELEMENT	36	37
41	42	43	44 B ¹⁰ GRADIENT ABSORBER	45	46	47
51	52 SM-2B	53	54 HIGH BURN UP CORE I ELEMENT	55	56	57
61	62	63	64	65 SM-2A	66	67
	72 IMPACT SAMPLES	73	74	75	76

Figure 3.1 SM-1 Core II, With Special Test Elements

7. Two reference SM-2 plates employing Gerdau UO_2 particles rather than spherical UO_2 . Fuel loading is 49.41 gm U-235 and 0.103 gm B-10.
8. CbB_2 poison has been substituted for ZrB_2 in two reference SM-2 plates. Fuel loading is 49.52 gm U-235 and 0.103 gm B-10.
9. The ZrB_2 poison in two reference SM-2 plates has been coated with Cb. Fuel loading is 49.45 gm U-235 and 0.101 gm B-10.

The SM-2B element was also used to spike SM-1 rearranged Core I. It will have received 3 percent fuel burnup and 15 percent boron burnup when it is inserted in Core II.

3.1.3 Testing Procedures, SM-2 Element

The SM-2A and SM-2B elements have been in SM-1 Core I for approximately 1.3 MWYR.

Miniature fuel plates that essentially duplicate each of the fuel plates have received extensive testing as part of the SM-2 program. ⁽⁷⁾ BM1 has cooperated in conducting this rigorous examination of these elements as part of this SM-2 program and their conclusions are published as part of the previous reference. Miniature specimens have received burnups of up to 46 percent uranium depletion reported by isotopic analysis and 56 percent reported by radiochemical analysis in the MTR and ETR capsules. Hot cell examination indicates excellent performance. Additional samples are now in the ETR and will receive approximately 65-90 percent burnup. The miniature plates operated at heat fluxes greater by a factor of 3 than those they will experience in the SM-1.

The SM-2 clad consists of 347 ss with low cobalt and tantalum content. The cobalt and tantalum exist in the clad as impurities. This report discusses the removal of these impurities elsewhere and indicates that no metallurgical properties of the clad are compromised by lowering the cobalt and tantalum content.

The welding assembly method is also unique to the SM-2 elements. The welds were tensile tested to 1400 lb per joint or about 70,000 psi with penetrations of 0.10 to 0.035 in., comparable to the strength of the base material.

To check the dimensional stability at reactor operating temperatures and the effect of temperature cycles, annealed and cold-rolled depleted elements were thermal cycled between room temperature and 600°F and 650°F at a rate of 100°F/hr. Maximum change in channel spacing was 0.002 inches. An element was heated to 800°F in a furnace, held 30 hours and furnace cooled. This is more than equivalent to design life in terms of relaxation of welding stresses.

The maximum change in channel spacing was 0.002 inches. A more accurate simulation of actual fuel element temperature profile was desired than that obtained by furnace heating because of the difference in temperature between core area and dead edge and side plate area in reactor operation. ⁽⁷⁾ This temperature difference was simulated in a test rig. This test also indicated very little change in channel dimensions.

Extensive corrosion tests, including Huey and static autoclave tests, were conducted and demonstrated that welds made by the reference method will provide integrity. ⁽⁷⁾ The ability of the reference welding process ⁽⁸⁾ to consistently produce sound welds has been demonstrated. ⁽⁹⁾

3.1.4 Summary

There is no significant increase in the hazard potential associated with SM-2 fuel elements. The operation of these fuel elements is extremely important prior to the operation of a full core of these elements.

3.2 TWO HIGH BURNUP CORE I STATIONARY FUEL ELEMENTS

After 16.4 MWYR in SM-1 Core I and an estimated additional 1.3 MWYR in the rearranged and spiked core, the standard SM-1 elements, numbered 80, 81 and 65, will have experienced about 46 a/o U-235 burnup and very close to 100 percent B-10 depletion. Evaluation of the effect of this extended burnup in Core II of SM-1 Core I fuel elements indicate several conceivable operational hazards, which are discussed below.

3.2.1 Irradiation Blistering

Numerous SM-1 type fuel plates have been irradiated to date, even at burnups approaching the maximum conceivable for these extended burnup elements and show no indication of failure. ⁽⁷⁾⁽¹⁰⁾ In addition, correlation with other methods and analysis ⁽¹¹⁾⁽¹²⁾ indicate that complete fissioning of the fuel will not generate sufficient gas pressure to cause blistering by matrix deformation.

3.2.2 Fuel Plate Rippling

Post-irradiation measurements of test elements in VBWR and SM-1 Core I elements removed after two-thirds core life indicate that rippling does increase under operation. ⁽¹³⁾⁽¹⁴⁾ However, the maximum ripple measured at ORNL for the two SM-1 fuel elements removed at 10.5 MWYR was + 13 mil from an arbitrary selected zero. ⁽¹⁴⁾ Because the heat flux is considerably less than that previously experienced, no major increase in ripple is expected.

Periscopic examination of the elements to be used will be made before insertion in Core II. If the rippling at that time is of such a magnitude that a projection of it indicates possibility that fuel plates will touch, that element will not be reinserted.

3. 2. 3 Loss of Comb Due to Braze Metal Corrosion

Although unalloyed braze metal is undoubtedly substantially removed, alloyed braze metal, the amount of which is a function of joint clearances, will be practically unaffected by corrosion. Periscopic examination of the attachment of these combs will be made before their insertion in Core II. If there is any doubt as to the integrity of this attachment, the element will not be used.

3. 2. 4 Loosening of Fuel Plate Due to Braze Metal Corrosion

Although a significant portion of the unalloyed braze metal will be removed, sufficient alloyed braze metal will remain to hold the fuel plates in place, although even in the absence of braze metal, the plates will not fall from the grooves. Periscopic examination of the outside joints will be made to verify the presence of sufficient alloyed braze metal prior to insertion of the elements.

3. 2. 5 Clad Cracks and Waterlogging

The occurrence of waterlogging requires the presence of void space sufficient to allow steam formation. Clad cracks observed in the elements removed after two-thirds of Core I life revealed excellent core to clad and frame bonding with no voids. ^(13, 14) A major change in fuel plate characteristics, contrary to past experience, would be required for waterlogging to occur.

3. 2. 6 Activity Increase from Fission Product Escape Through Clad Cracks

It was observed⁽¹³⁾ that there were a number of penetrations through the cladding at grain boundaries in the two Core I elements removed at 10.5 MWYR. These micro-cracks result in escape by recoil, and diffusion of fission products into the water and have resulted in the low fission product activity of SM-1 water. The coolant activity for Core II will be much less than that previously experienced because only two of these SM-1 Core I elements will be inserted with 43 unirradiated components. The activity in the system to date has been reported to have no adverse effects upon operation.

3. 2. 7 Summary

There is no significant increase in the hazard potential with the insertion of the high burnup Core I elements in Core II. Extended burnup data on

the UO_2 stainless steel dispersion fuel system, at operating conditions not available in any other test work, will be made available by these elements. This information will provide in turn a sounder basis for correlation of fuel element performance than is now available.

3.3 SM-1 CORE II MATERIALS TEST PACKAGE

3.3.1 Description

The dummy fuel elements containing test materials to be irradiated in Core II will be placed in element positions 12 and 72. Each dummy element will contain twelve Naval Research Laboratory fabricated capsules (of the type shown in Fig. 3.2) in four rows, each row consisting of three capsules placed end to end. Each capsule will contain fifteen standard V-notch Charpy impact test specimens arranged five specimens wide by three specimens long by one specimen thick in the capsule shown in Fig. 3.3. In all, 360 specimens will be placed in the dummy elements.

3.3.2 Impact Specimens

The specimens are standard V-notch Charpys prepared in accordance with requirements of ASTM E-23-56T Type A specification. Each specimen is 0.394 in. (10 mm) square by 2.165 in. (55 mm) long, with a transverse notch at the mid-length. Three general types of steels will be tested:

Carbon steel	ASTM A-212 Grade B
Alloy steels	ASTM A-350 Grade LF-1 Modified ASTM A-350 Grade LF-3 ASTM A-393 ASTM A-302B
Austenitic stainless	Type 304 Type 347

3.3.3 Flux Monitors

Four different non-fission threshold detector type monitors⁽¹⁵⁾ are enclosed in each material test capsule. These activation monitors, and their respective energy thresholds, are:

1. Cobalt	2.5×10^{-8} mev
2. Cadmium - shielded cobalt	1.2×10^{-4} mev
3. Sulfur	2.9 mev
4. Nickel	5.0 mev

Figure 3.3 Fuel Element Stationary - Dummy

Figure 3.4 Capsules for Irradiation Experiment

Figure 3.5 Radiation Exposure Container

The first three are separately sealed in quartz capsules. These quartz capsules are placed in slotted recesses machined into the side of the stainless steel capsule frame. The nickel monitor is in the form of wire, which is laid into the notch of the impact specimens.

3.3.4 Temperature Indicators

The approximate maximum temperature attained by the specimens during irradiation will be indicated by a series of low melting metals or metal alloys contained within each capsule.

Each indicator consists of a foil strip 0.005 to 0.010 in. thick by 1/16 in. wide by 3/8 in. long. These indicators weigh approximately 50 mg each. The melting point and composition of the indicators are:

450 ⁰ F	Tin
475 ⁰ F	83 Pb - 17 Cd
486 ⁰ F	88.8 Pb - 11.2 Sb
510 ⁰ F	82.5 Cd - 17.5 Zn
573 ⁰ F	97 Sn - 3 Cu

The temperature indicators are contained within sealed quartz capsules placed in machined recesses inside the capsule frame.

3.3.5 Fabrication

The capsules, to be fabricated by NRL, will be welded around the periphery, and helium leak tested under water subjected to a pressure of 1500 psia. The breather tube will then be welded closed. This technique has undergone many successful tests. ⁽¹⁶⁾ At the MTR, an experiment placed in the reflector involved six drop-weight test specimens (~ 800 gm each), twenty Charpy V-notch specimens (~ 40 gm each), and two tensile specimens (~ 15 gm each) placed in the reflector, where gamma heating was 2 to 4 watts per gram in this position. The total weight of the assembly was 6000 grams. More than 20 of these assemblies have been successfully irradiated at temperatures below 300⁰F in a position where approximately 20,000 watts of heat were generated in the capsule. In the core of the LITR, several different types of irradiation experiments were performed using the same techniques as in the MTR assemblies. ⁽¹⁷⁾

3.3.6 Justification for Insertion in Core II

The only conceivable significant hazard involved would be a sudden loss of these specimens, creating in effect a reactivity increase. This possibility is very remote due to the nature of the design. The thermal section of this report indicates temperature far below the melting temperature and the capsules cannot fall through the tube; their physical dimensions will not allow it. The screws holding the plates cannot physically come out and the screws joining the side plates to the end box castings are each staked in three places. Irradiation of these impact specimens is of vital significance in obtaining data for the continued safe operation of the Army reactors. These impact specimens will undergo rapid neutron exposure, the equivalent of twenty years of plant operation, in about three years. This will provide invaluable data on the variation of nil ductility temperatures with fast nvt.

3.4 PM-1-M FUEL ELEMENT

An addendum to this report will be issued to cover this element. This is necessary because the final design of this element was not available in time to be included in this report.

3.5 BORON-GRADIENT ABSORBER⁽¹⁸⁾

Irradiation stability of the stainless steel clad iron-boron absorbers used with Core I of the SM-1 reactor was limited by the fact that boron burnup was concentrated at the core-clad interface of the composite absorber rather than dispersed uniformly through the thickness of the plate. This inordinate local accumulation of damage was a result of the large self-shielding effects associated with a high concentration (2.8 w/o) of B-10, which has a thermal neutron cross section of 4020 barns.

Post-irradiation examination of two of these absorber sections removed from the SM-1 after an exposure of 10.5 MWYR indicated the existence of clad failures and core-clad bond separations in those regions near the absorber tips which had received the highest burnups. Bond separation extended, in the worse case, to a point approximately 7 in. from the tip of the absorber where the estimated burnup was 6 a/o B-10. ⁽¹³⁾

Based on these observations and observations on similar miniature absorber sections irradiated in MTR process water at 200°F to 3.8 a/o B-10 burnup with no damage, ⁽¹⁹⁾ the boron-gradient absorber concept was formulated. ⁽²⁰⁾ In this concept, the boron is dispersed through the thickness of the absorber section in such a graduated manner that the relative amount of boron burnup occurring in each increment of thickness is essentially the same.

Further, the absolute amount of boron burned in each incremental layer is limited to that which the cited experiments have shown to be acceptable from the standpoint of structural damage. For example, the boron burnup at the core-clad interface of the miniature absorber, which received a thickness-averaged burnup of 3.8 a/o, was 35 a/o. Expressed differently, the boron content at the interface was reduced from 2.86 w/o to 1.86 w/o. It would thus appear that about 1 w/o burnup can be tolerated without gross damage. A similar case can be made using results from the full-size absorbers irradiated in SM-1. Therefore, the boron gradient rod was designed in a manner so that a maximum of 1 w/o B-10 will be burned uniformly throughout the rod during in-pile exposure of 6.1×10^{20} nvt. Assuming an average thermal neutron flux of 1.36×10^{13} nv at the rod tip⁽²¹⁾ and an average power level of 10 MW, this is equivalent to 14 MWYR of operation. Since Core II or SM-1 is scheduled for 12 MWYR operation, a margin of conservatism exists.

In the event that the gradient absorber receives a higher exposure than indicated above, the additional burnup will be concentrated at the center rather than near the surface of the rod. This should tend to minimize damage. In addition, the matrix material for this gradient absorber is a Type 200, austenitic stainless steel* which should have superior corrosion resistance to the elemental iron matrix in the original absorbers, should a clad failure occur.

Experience with the Fe-B absorbers in SM-1 indicates that even if clad fracture or bond separation occurs, the consequences of such an event should not be serious. The SM-1 control rod function was not affected by the generation of bond failures and no significant problems were encountered in removing the absorber sections from the core for post-irradiation examination.

Boron loadings and concentrations through the absorber thickness are presented in Table 3.1.

3.5.1 Summary

It is considered that this rod will perform at least as well as the boron rods originally in the SM-1 and will contribute to the development of a more economical absorber. Accordingly, it is important in the study of replacement cores.

* Nominal Composition - 0.03% C; 0.06% Si; 20.7% Mn; 0.008% S; 0.009% P; 16.37% Cr; 0.49% Ni; bal Fe.

TABLE 3. 1
BORON LOADING IN EXPERIMENTAL ABSORBER BG-I

<u>Type of Core</u>	<u>B-10 Concentration (w/o)</u>	<u>B-10 Content (gm)</u>
1 Surface	1. 04	0. 70
2	1. 13	0. 52
3	2. 27	0. 47
4	3. 20	3. 48
Center		
4	3. 20	3. 48
3	2. 27	0. 47
2	1. 13	0. 52
1 Surface	1. 04	0. 70

4.0 NUCLEAR CHARACTERISTICS

4.1 INTRODUCTION

The fuel and poison loadings in the standard SM-1 Core II elements are essentially identical to SM-1 Core I elements. Material composition of Core II Components are given in Table 4. 1. The measured nuclear characteristics are essentially identical. (2) The inclusion of the special components in SM-1 Core II results in a reduction in reactivity and change in power distribution. Analytical predictions of these changes have been made. (4) In addition, the effect of these special components have been measured in a zero power experiment. (4) Only part of the final experimental data was available at the time this report was prepared. Complete results will be reported in Reference 4.

4.2 CORE REACTIVITY

At start of life, the reactivity of SM-1 Core I was 15.4% ρ . Changes in fuel and poison loadings in the basic Core II elements decreases the worth to 14.6% ρ . The worth of Core II in its final operating configuration will be further decreased by about 3.0% ρ due to insertion of special test assemblies. Hence, at start of life, SM-1 Core II will have 11.6% ρ reactivity.

4.3 CONTROL ROD WORTH

The five rod bank in SM-1 Core II will include four Eu_2O_3 absorber sections and one gradiated boron section. The four Eu_2O_3 rods were used for about 8 MWYR in SM-1 Core I and SM-1 rearranged and spiked Core I. During this operation, the rods experienced an average burnup of 0.03 absorptions per original Eu atom. The decrease in rod worth associated with a burnup of this magnitude is negligible. The gradiated boron absorber contains 10 gm B-10. This rod has been found to be worth 5 cents more than a Eu_2O_3 rod; (4) hence, the absorber sections in the 5 rod bank will be worth essentially the same as those used in SM-1 after 10.5 MWYR.

The two boron absorber sections in positions A and B in Core I will be replaced in Core II by Eu_2O_3 absorber sections. The reactivity effect associated with replacing the boron absorbers by Eu_2O_3 is a loss in shut-down reactivity of about 40 cents.

Stuck rod measurements were made in the zero power experiment for SM-1 Core II with mocked up special elements. Preliminary results are given in Table 4. 2. A correction of +0.163 in. must be added to the rod position to get dial readings.

TABLE 4.1
MATERIAL COMPOSITION OF CORE II COMPONENTS

	<u>Fuel Elements</u>		
	<u>Number in Core</u>	<u>U-235 (gm)</u>	<u>B-10 (gm)</u>
Standard SM-1 Core II Stationary Element	31	515.16	.4163
Standard SM-1 Core II C. R. Fuel Element	7	401.12	.2813
SM-2A	1	868.7	1.54
SM-2B	1	863.7	1.35
High Burnup Core I Element	2	278	.011
PM-1-M	1		

	<u>Non Fuel Elements</u>	
	<u>Number in Core</u>	<u>Poison (gm)</u>
Eu ₂ O ₃	6	124 (Eu)
Gradiated Boron Steel Absorber	1	40 (B-10)
Impact Specimens	2	0

TABLE 4.2
STUCK ROD MEASUREMENTS IN SM-1 CORE II WITH
SPECIAL COMPONENT MOCKUPS

<u>CASE</u>	<u>RODS OUT</u>	<u>CRITICAL ROD</u>	<u>CRITICAL POSITION</u>
1	A & 1	2	5.160
2	A & 1	3	10.408
3	A & 1	4	4.606
4	A & 1	C	4.295
5	A & 1	B	9.200
6	A & 2	C	12.670
7	A & 2	4	13.234
8	A, 2 & 3		SUBCRITICAL
9	A & 2	B	10.358
10	A	4	18.129
11	A, B, & C		SUBCRITICAL
12	A, 3 & C		SUBCRITICAL
13	A, C	4	10.124
14	A & 3	4	12.192
15	A & B*	4	15.457
16	A & B*	3	18.230
17	A & B*	1	17.179
18	A & B*	2	13.112
19	A & C	4	10.132

* B at 19 in.

4.4 LOCAL POWER PEAKING

A consistent analysis of the radial power distributions in SM-1 Core I, SM-1 rearranged and spiked Core I, and SM-1 Core II has been made. (4) Local power peaking was evaluated on an element by element basis in all three cases. Comparison of the peaking factors in the three cases shows that maximum power generation occurs in the SM-2 element in position 65 of the rearranged and spiked Core I. Hence, the power peaking associated with SM-2 elements in Core II will be less severe than that already experienced in the rearranged and spiked SM-1 Core I. The most severe power peaking occurring in Core II will also be in the SM-2 element in position 65. A comparison of local power peaking in some of the hotter elements in SM-1 Core I, SM-1 rearranged and spiked Core I, and SM-1 Core II is given in the following tabulation.

Core Position	Core I		Rearranged and Spiked Core I		Core II	
	$Q(\Delta T)$	$Q(\Delta \theta)$	$Q(\Delta T)$	$Q(\Delta \theta)$	$Q(\Delta T)$	$Q(\Delta \theta)$
44	1.48	1.58	1.97	2.08	1.55	1.74
52	1.23	1.35	1.81	2.90	1.29	1.90
65	1.16	1.38	2.39	2.96	1.58	2.28
33	1.25	1.45	1.15	1.25	1.36	1.50
45	1.46	1.57	1.82	1.95	1.42	1.75

where,

$$Q(\Delta \theta) = P(x, y)_{hp} \max. F_{rods} F_{local}$$

and $P(x, y)_{\max.}$ = maximum local heat generation rate in each element

F_{rods}, F_{local} = experimental correction factors

$$Q(\Delta T) = P(x, y)_{hp} F_{rods} F_{local}$$

$P(x, y)_{hp}$ = calculated average heat generation rate in the hot plate of each element.

4.5 CONCLUSIONS

The insertion of the special core components in SM-1 Core II will result in a loss in reactivity of at least 3%. Therefore, stuck rod safety and shutdown margin are considerably improved over SM-1 Core I.

5.0 THERMAL CHARACTERISTICS

5.1 MINIMUM BURNOUT RATIOS

To verify the thermal safety of Core II, the elements with the highest power generation, namely, elements SM-2A in core position 65 and SM-2B in core position 52 were analyzed. In addition, the elements in positions 25, 45, 53, 67, 33, 44, and 55, which have high power to flow ratios, were analyzed.

The thermal analysis⁽⁴⁾ of these elements has been made by STDY-3⁽²⁾, an IBM-704 digital computer code. Performance of these elements is measured by the thermal characteristics of their "hot channels" which are postulated to have the highest element flux coupled with the most adverse deviations. Calculation of the nominal channels' bulk enthalpy and temperature rise and two-phase pressure drop are also made. Since both the nominal and hot channel are in parallel and have the same pressure drop, the hot channel flow is computed by comparison to the nominal channel while including effects such as plate to plate maldistribution. Once the hot channel flow is determined, bulk enthalpy, bulk temperature, surface temperature, and minimum DNB (departure from nucleate boiling) ratios are evaluated.

5.2 INPUT PARAMETERS

The important parameters which were used in formulating the input to the code can be categorized as hot channel factors and system parameters. Hot channel factors used in this analysis include local and average mechanical and metallurgical factors, nuclear uncertainty factors, instrumentation factors, and channel to channel maldistribution. Actual factors for each of the different type elements considered in this report are given in Table 5.1.

The system parameters include the element flow rates, average radial power peaking for the nominal $Q(\Delta T)$ and hot channels $Q(\Delta T)$ and the local radial power peaking $Q(\Delta \theta)$. Actual factors for each of the different elements considered in this analysis are tabulated in Table 5.2.

TABLE 5.1
SUMMARY OF HOT CHANNEL FACTORS

Type of Deviation	Deviation		F _{av}	F _{local}
Active Core Length				
a. SM-1 Core I stationary element	22"	+ 1/2" - 1"	1.0357	-
b. SM-1 Core I control rod	22"	+ 1/2"	1.0233	-
c. SM-1 Core II stationary element	21-3/4"	+ 3/4"	1.0357	-
d. SM-1 Core II control rod	21-1/8"	+ 5/8"	1.0299	-
e. SM-2 stationary element	22"	+ 1/2"	1.0233	-
Uranium Content (all elements)			1.005	1.025
Clad Thickness (all elements)	.005"	+ .0003"	1.007	-
		Av		
	.005"	+ .005"	-	1.012
		loc		
Nuclear Uncertainty (all elements)			1.05	1.10
Total Reactor Power Generated in Fuel Plates (all elements)			1.000	0.950
Instrumentation Tolerances (all elements)				
a. System pressure	+ 25 psi at 1200 psi = 1175 psi (min.)			
b. Core power	+ 3.5% at 13 MW = 13.45 MW(max)			
c. Inlet temperature	+ 4° at 427.7°F = 431.7°F (max.)			
Channel Spacing (Inclusive of hot rippling)	Av Dimension	Min. Av Dimension	Max Local Dimension	
a. SM-1 Core I & II stationary elements	0.133"	0.128"	0.172"	
control rod	0.133"	0.128"	0.172"	
b. SM-2 stationary element	0.123"	0.119"	0.136"	
Plate to Plate Maldistribution (all elements)	12% mass flow			

TABLE 5.2
SUMMARY OF BASIC INPUT PARAMETERS FOR
THE ELEMENTS INVESTIGATED

<u>Core Position</u>	<u>Type of Element</u>	<u>Nominal Channel Flow Rate G x 10⁵ lb/hr-ft²</u>	<u>Q(ΔT)</u>	<u>Q(ΔT)</u>	<u>Q($\Delta \theta$)</u>
25	SM-1 Core II Stationary	8.695	.940	1.25	1.55
45	SM-1 Core II Stationary	10.358	1.293	1.42	1.75
53	SM-1 Core II Stationary	9.593	1.094	1.52	1.65
67	SM-1 Core II Stationary	4.453	.656	1.09	1.22
33	SM-1 Core II Control Rod	9.555	1.135	1.36	1.50
44	SM-1 Core II Control Rod	10.105	1.397	1.55	1.74
55	SM-1 Core II Control Rod	9.555	1.105	1.58	1.75
52	SM-2 Stationary	8.934	.652	1.29	1.90
65	SM-2 Stationary	8.985	.869	1.58	2.28
44	SM-1 Core I Stationary	10.105	1.380	1.48	1.651

5.3 STEADY STATE ANALYSIS

To insure the conservatism of the analysis, all possible instrument and mechanical tolerances or uncertainties have been taken into account. The two power levels that were investigated are the nominal of 10.77 MW as is reported in APAE 18⁽²³⁾ and the maximum overload that is possible without a reactor scram of 13.45 MW.

The results of the analysis of the special Core II elements are tabulated in Table 5-3. The calculated plate surface temperatures for the two controlling elements in Core II, namely, the SM-2 elements in core positions 52 and 65, are also plotted in Figures 5.1 and 5.2 for the two power levels. The more severe case of 13.45 MW indicates that the element in position 65 has 11.5°F of superheat for approximately 8 in. while the element in position 52 indicates 11°F for 3 inches. This gives the appearance of nucleate boiling, but an investigation of these elements on an average or bulk basis and a comparison to the Jens and Lottes⁽²²⁾ criteria for the superheat necessary to cause nucleate boiling indicates

that there is no boiling and associated flow readjustment in these elements. Furthermore, similar evaluations on all the elements analyzed by the STDY-3 code indicates that none of them are in nucleate boiling.

TABLE 5.3
SUMMARY OF RESULTS OF STEADY STATE ANALYSIS

Core Position	Core No.	Core Power MW	Nominal Channel Flow $G \times 10^5$, lb/hr ft ²	Hot Channel Flow $G \times 10^5$, lb/hr ft ²	Hot Channel Bulk Outlet Temperature, °F	Hot Channel Maximum Plate Surface Temperature, °F	DNBR
25	II	10.77	8.695	7.608	453.9	544.97	11.5
45	II	"	10.360	9.065	468.7	542.11	10.2
53	II	"	9.593	8.394	474.4	545.01	10.7
67*	II	"	4.453	4.175	479.3	575.21	14.2
33	II	"	9.555	8.361	455.8	534.28	11.9
44	II	"	10.110	8.846	457.5	544.77	10.2
55	II	"	9.555	8.361	459.6	551.08	10.3
52	II	"	8.934	7.817	473.0	558.50	9.4
65*	II	"	8.985	7.862	482.0	576.16	7.8
25	II	13.45	8.695	7.608	477.5	570.25	9.1
45	II	"	10.360	9.065	475.1	566.71	8.1
53	II	"	9.593	8.394	482.1	570.26	8.6
67*	II	"	4.453	4.175	504.9	575.78	11.3
33	II	"	9.555	8.361	459.1	556.93	9.5
44	II	"	10.110	8.846	461.2	569.98	8.2
55	II	"	9.555	8.361	463.8	576.78**	8.1
52	II	"	8.934	7.817	480.5	576.20	7.5
65*	II	"	8.985	7.862	491.5	576.78	6.2

* These elements do not indicate any flow readjustment due to nucleate boiling but they do indicate a very slight margin from the inception of nucleate boiling.

** This control rod element reaches a peak value of 576.78°F at only one point and falls off rapidly.

Figure 5.1 Element Nos. 65 and 52, Hot Channel Plate Surface Temperature Vs. Length Along Plate, 13.45 MW

Figure 5.2 Element Nos. 65 and 52, Hot Channel Plate Surface Temperature Vs. Length Along Plate, 10.77 MW

5.4 CONCLUSIONS

It is concluded from the tabulation of results of the steady state analysis (Table 5.3) that no Core II element is in danger of burnout, since the most critical element, the SM-2 stationary element in core position 65, has a minimum DNBR of 7.8 at 10.77 MW and 6.2 at 13.45 MW. The margin of safety with burnout ratios of this magnitude can be seen by comparison with the minimum allowable design DNBR of 2.5⁽²⁴⁾ in steady state and 2.0 in a transient.

Since there is no nucleate boiling indicated in the critical elements analyzed, and the minimum DNBR is 2.5 times greater than an accepted design criteria, the entire SM-1 Core II will be safe thermally under steady state conditions.

5.5 TRANSIENT ANALYSIS

5.5.1 Discussion

A pseudo-transient analysis has been formulated with the steady state STDY-3 code by making consecutive runs where both the power and flow rate are known as a function of time from experimental data.⁽²⁵⁾ This data is based on a long delay in scram on low flow. This has been performed only on the most critical element, which is the SM-2 element in core position 65, because if this element is safe during a loss of flow accident, Core II, as a whole, will be thermally safe under this condition. To obtain the severest condition, the analysis has been made for core powers of 10.77 and 13.45 MW.

Results of this analysis, inclusive of the steady state (zero time) are tabulated in Table 5.4, and the minimum burnout ratio is plotted in Fig. 5.3. The DNBR has been carried to three decimal places only to show the magnitude of its change during the early periods of the transient. Scram occurs 0.14 sec after pump failure.

5.5.2 Conclusions

The results of the transient analysis of element 65, shown in Table 5.4 and Fig. 5.3, indicate only a slight decrease in DNBR at 0.05 sec coinciding with a very slight increase in plate surface temperature. After this time period, the power falls off at a sufficiently fast rate in comparison to the flow, so that the minimum DNBR increases and the maximum plate surface temperature decreases. Actually, the reactor will scram after 0.14 sec. With relatively high minimum DNBR's of 6.15 in the 13.45 MW transient and 7.73 in the 10.77 MW transient, the critical element 65 has a considerable margin of safety from burnout when compared to the aforementioned criteria. As in the steady state case, there is no nucleate boiling noted at both power levels, but once more the inception of boiling is very marginal. Therefore, it is concluded that element 65 is safe in a transient and Core II will be safe during the loss of flow accident.

Figure 5.3 Burnout Ratio Vs. Time For Element No. 65

TABLE 5.4
TRANSIENT ANALYSES OF STATIONARY ELEMENT 65 FOR
INITIAL CORE POWERS OF 10.77 MW AND 13.45 MW

Time (Sec.)	Core Power (MW)	Nominal Channel Flow (G x 10 ⁵ lb/hr-ft ²)	Hot Channel Flow (G x 10 ⁵ lb/hr-ft ²)	Hot Channel Bulk Outlet Temperature (°F)	Hot Channel Max Plate Surface Temp. (°F)	Nucleate Boiling	Quality	DNBR
0	10.77	8.985	7.862	482.0	576.16	No	None	7.754
.05	10.76	8.266	7.233	486.2	576.26	"	"	7.732
.10	10.69	7.952	6.958	488.0	576.25	"	"	7.767
.15	10.59	7.476	6.541	490.8	576.22	"	"	7.825
.20	10.46	7.188	6.289	492.4	576.18	"	"	7.910
.30	10.25	6.649	5.818	495.9	576.13	"	"	8.042
.40	10.04	6.218	5.441	498.8	576.07	"	"	8.193
.60	9.66	5.661	5.307	498.0	575.97	"	"	8.520
0	13.45	8.985	7.862	491.5	576.78	No	None	6.171
.05	13.44	8.266	7.233	496.7	576.90	"	"	6.148
.10	13.36	7.952	6.958	498.9	576.88	"	"	6.173
.15	13.22	7.476	6.541	502.5	576.85	"	"	6.215
.20	13.06	7.188	6.289	504.4	576.81	"	"	6.281
.30	12.80	6.649	5.818	508.7	576.76	"	"	6.381
.40	12.54	6.218	5.829	507.0	576.70	"	"	6.529
.60	12.06	5.661	5.307	511.2	576.59	"	"	6.751

6.0 MAXIMUM TEMPERATURE IN MATERIALS TEST PACKAGES

6.1 TEMPERATURE DETERMINATION

A mass flow rate through the impact specimen of 6.198×10^5 lb/hr-ft² was calculated at bulk water conditions of 440°F and 1200 psia. Using the Dittus-Boelter equation, a film coefficient on 2000 Btu/hr-ft²-°F was calculated on the surface of the impact sample capsule. For the purpose of conservatism, the maximum gamma heat generation of 1260 Btu/hr-in³ (22.3 watt/cc) which exists approximately 11 in. from the bottom of the core is assumed in the calculations. The calculated film drop is 19°F.

The contact resistance is negligible between impact sample and cladding, (26) using type 347 stainless steel properties, the internal temperature rise is 30°F. Using a conservative estimate of a 5°F rise in bulk water temperature, the maximum internal temperature is:

$$433 + 5 + 19 + 30 = 487^\circ\text{F}$$

This is determined by a one-dimensional analysis. A two-dimensional analysis would be more realistic and produce a lower maximum temperature. However, for the purposes of establishing the safety of the impact specimen, it is concluded that a 49°F temperature rise over bulk water temperature is the maximum that could be expected.

6.2 CONCLUSION

A theoretical analysis indicates that the maximum internal temperature of the impact capsules will be 487°F. This temperature is low enough to be of no concern. In addition, NRL has extensive satisfactory experience with this type of capsule at gamma heating rates of 2 to 4 watts/gm. This compares with a maximum value of 2.9 watts/gm gamma heating rate in SM-1.

The impact sample will perform satisfactorily from the thermal and hydraulic aspect, and will have no adverse affect on SM-1 Core II performance.

7.0 DISCUSSION AND CONCLUSIONS

The operation of this core does not represent any additional hazard to the SM-1. It does not introduce any new hazard nor does it add to the magnitude of any hazard previously considered. No loss of training time or other expected use of the SM-1 is expected due to the operation of Core II with the special components. This, coupled with the major advantages of determining technological factors needed in the growth of the Army Nuclear Power Program, calls for the insertion of this core with its special components.

Loading and startup procedures for the core will be determined in the Alco Critical Facility prior to the insertion of Core II in the SM-1.

BIBLIOGRAPHY

1. Rosen, S.S., "Hazards Summary Report for the Army Package Power Reactor SM-1 Task XVII," APAE 2, Rev. 1, May, 1960.
2. Robinson, R.A., et al, "SM-1 Research and Development Task XV Zero Power Experiments for SM-1 Core II and SM-1A Core I," APAE 58 Revised, October 9, 1960.
3. Lee, D.H., Marimpietri, A., "Evaluation of Special Components Suggested for Insertion in SM-1 Core II," AP Note 281, Alco Products, Inc., November, 1960.
4. Lee, D.H., et al, "Experiments and Analysis for SM-1 Core II," APAE No. 85, to be published.
5. Leitten, C.D., Jr., "The Irradiation of Minature Eu_2O_3 - Bearing Absorber Plates," Oak Ridge National Laboratory, Irradiation Request ORNL-MTR-39, ORNL CF-59-9-25, September, 1959.
6. Frye, J.H., Jr., et al, "Metallurgy Division Annual Progress Report for Period Ending July 1, 1960," ORNL-2988, September, 1960.
7. Hoover, H.L., Project Engineer, "SM-2 Core and Vessel Design Analysis," APAE No. 69, Vol. II, March 7, 1961.
8. Alco Specification F-31, "Fuel Elements, SM-2 Nuclear Power Plant."
9. Harris, R.L., "SM-2 Fuel Element Welding Development, Task 5.0," APAE No. 70, August, 1960.
10. ORNL CF 58-2-71, Classified.
11. Weir, J.R., "A Failure Analysis for the Low Temperature Performance of Dispersion Fuel Elements," ORNL 2902.
12. Keller, D.L., et al, "A Method of Correlating Irradiation Effects in Dispersion Fuel," Battelle Memorial Institute, BMI 1408.
13. Army Package Power Reactor Project Annual Progress Report for Period Ending January 31, 1960, ORNL-2907.
14. Letter Communication to J.G. Gallagher, Alco, from W.C. Thurber, Metallurgy Division, Oak Ridge National Laboratory, June 3, 1960.

15. Steele, L. E., and Hawthorne, J. R., "Neutron Flux Measurements for Materials Irradiation Experiments at Argonne National Laboratory, Brookhaven National Laboratory, Oak Ridge National Laboratory, and National Reactor Test Station," Naval Research Laboratory, NRL 5483, May, 1960.
16. Hawthorne, J. R., and Steele, L. E., "The Effect of Neutron Irradiation on the Charpy V and Drop Weight Test Transition Temperatures of Various Steels and Weld Metals," NRL 5479, 1960.
17. Letter communciation of L. E. Steele U.S. Naval Laboratory, March 3, 1961.
18. Letter communciation to J. G. Gallagher from W. C. Thruber, Oak Ridge National Laboratory, March 20, 1961.
19. Richt, A. E., "Irradiation Stability of APPR-1 Type Fe-B Cored, Stainless Steel Clad Control Rod Specimens," ORNL-CF-58-5-60, May 14, 1958.
20. Watts, T. D., et al, "Boron Gradient for Control Rod Application," in Metallurgy Division Annual Progress Report, ORNL 2988, September 9, 1960.
21. McElligott, P. E., "Burnout Distribution in SM-1 Control Rod Elements, Fixed Element #57 and Absorber Sections at 10.5 MWYR," APAE Memo No. 199, June 15, 1959.
22. Pyle, R. S., "A Program for the Thermal Analysis of a Pressurized Water Nuclear Reactor During Steady-State Operation," Westinghouse Electric Co., WAPD-TM-213, June 1960.
23. Meem, J. L., "Initial Operation and Testing of the Army Package Power Reactor APPR-1," APAE 18, August 1957.
24. Hoover, H. L., Project Engineer, "SM-2 Core and Vessel Design Analysis," APAE No. 69, Vol. I, March, 1960.
25. Morrison, J. H., Lee, D. H., "Evaluation of Loss of Flow Accident in the SM-1," Alco Final Report Test 600, April, 1960.
26. Graff, W. J., "Thermal Conductances Across Metal Joints," Machine Design V. 32, No. 19, September 15, 1960.

END