SEP 5 1961

MASTER

Copy No 19

APAE Memo - 322

AEC Research and
Development Report
UC-81, Reactors - Power
[Special Distribution]

MASTER

**correlation of critical mass data
on light water moderated, fully
enriched uranium, stainless
steel reactors, part I**

Contract No. AT[30-1]-2639
with U. S. Atomic Energy Commission
New York Operations Office

ALCO PRODUCTS, INC.
NUCLEAR POWER ENGINEERING DEPARTMENT

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

APAE Memo-322

AEC Research and
Development Report
UC-81, Reactors, Power
(Special Distribution)

**CORRELATION OF CRITICAL MASS DATA
ON LIGHT WATER MODERATED,
FULLY ENRICHED URANIUM,
STAINLESS STEEL REACTORS
PART I**

By
D. H. Lee

Approved By:
E. W. Schrader, Project Engineer

Issued: July 17, 1962

Contract No. AT(30-1)-2639
With U.S. Atomic Energy Commission
New York Operations Office

ALCO PRODUCTS, INC.
Nuclear Power Engineering Department
Post Office Box 414
Schenectady 1, New York

AEC LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights:

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

ALCO LEGAL NOTICE

This report was prepared by Alco Products, Incorporated in the course of work under, or in connection with, Contract No. AT(30-1)-2639, issued by U. S. Atomic Energy Commission, NYOO; and subject only to the rights of the United States, under the provisions of this contract, Alco Products, Incorporated makes no warranty or representation, express or implied, and shall have no liability with respect to this report or any of its contents or with respect to the use thereof or with respect to whether any such use will infringe the rights of others.

DISTRIBUTION

External Copies

- 1-2 New York Operations Office
 U. S. Atomic Energy Commission
 376 Hudson Street
 New York 14, New York

 Attn: I. M. Adler
- 3 New York Operations Office
 U. S. Atomic Energy Commission
 376 Hudson Street
 New York 14, New York

 Attn: Library
- 4-6 U. S. Atomic Energy Commission
 Washington 25, D. C.

 Attn: Chief, Water Systems Project
 Branch (Army Reactors)
 Division of Reactor Development
 Mail Station F-311
- 7 U. S. Atomic Energy Commission
 Washington 25, D. C.

 Attn: Chief, Evaluation and
 Planning Branch
 Civilian Reactors
 Div. of Reactor Development
 Mail Station F-311
- 8 U. S. Atomic Energy Commission
 Chief, New York Patent Group
 Brookhaven National Laboratory
 Upton, New York

 Attn: Harman Potter

External
Copies

DISTRIBUTION (CONT'D)

- 9 Idaho Nuclear Power Field Office
U. S. Army Engineer Reactors Group
P. O. Box 2108
Idaho Falls, Idaho
- 10 U. S. Atomic Energy Commission
Reports and Statistics Branch
Division of Reactor Development
Washington 25, D. C.
- 11-12 Office of the Chief of Engineers
Department of the Army
Building T-7
Washington 25, D. C.

Attn: Chief, Projects Branch
Nuclear Power Division
- 13-15 Nuclear Power Field Office
U. S. Army Engineer Reactors Group
Fort Belvoir, Virginia

Attn: Chief, Nuclear Power
Field Office
- 16 Nuclear Power Field Office
U. S. Army Engineer Reactors Group
Fort Belvoir, Virginia

Attn: O. I. C. SM-1
- 17 Chief, U. S. Army Reactors Group
Fort Greely, Alaska
APO 733
Seattle, Washington

Attn: O. I. C. SM-1A
- 18 Commanding Officer
U. S. Army Polar Research
and Development Center
Fort Belvoir, Virginia

Attn: Nuclear Power Officer

External
Copies

DISTRIBUTION (CONT'D)

19-21 Office of Technical Information Extension
 P. O. Box 62
 Oak Ridge, Tennessee

22 Union Carbide Nuclear Corporation
 Oak Ridge National Laboratory
 Y-12 Building 9704-1
 P. O. Box "Y"
 Oak Ridge, Tennessee

 Attn: L. D. Schaffer

23 The Martin Company
 P. O. Box 5042
 Middle River, Maryland

 Attn: AEC Contract Document Custodian

24-25 Combustion Engineering, Incorporated
 Nuclear Division
 Prospect Hill Road
 Windsor, Connecticut

 Attn: Mr. J. B. Anderson

26 Alco Products, Inc.
 P. O. Box 145
 Fort Belvoir, Virginia

 Attn: W. C. Best

Internal
Copies

27	K. Kasschau	34-35	D. H. Lee (2)
28	J. G. Gallagher	36	F. G. Moote
29	J. F. Haines	37	T. M. Raby
30	R. Bell	38	L. D. Walthousen
31	C. Klotz	39	E. W. Schrader
32	W. T. Williams	40	R. E. Neville
33	P. E. Bobe	41-45	NPED file

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

ABSTRACT

Experimental data has been collected on over 70 light water moderated, fully enriched uranium, stainless steel, critical cores. An equation for the critical mass of cores with a buckling of 0.007 cm^{-2} that is linearly dependent on stainless steel volume fraction and grams of B-10 has been compared with available critical experiments and found to yield reasonable results. A correlation method, relating buckling to $\left(\frac{\xi \Sigma_s}{\Sigma_a} \right)$ has been found to fit the available experiments.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	vii
1.0 SUMMARY	1
2.0 INTRODUCTION	3
3.0 PREVIOUS CORRELATION ATTEMPTS	5
3.1 Experiments on SM-1 and SM-1 Mockups	5
3.2 Correlation of 7 x 7 Arrays of APPR - Type Elements	5
3.3 Discussion of Linear Correlation Model	8
4.0 CRITICAL MASS CORRELATION FOR BARE HOMOGENEOUS REACTOR	11
5.0 CALCULATIONAL TECHNIQUES	19
5.1 Equivalent Homogeneous Systems	19
5.1.1 Cell Calculations	19
5.1.2 Core Calculations	20
5.2 Equivalent Bare Cores	20
6.0 CORRELATION OF FLUX SPECTRA	23
7.0 CORRELATION OF CRITICAL BUCKLING	25
7.1 Discussion of Buckling Correlation	26
8.0 CONCLUSIONS AND RECOMMENDATIONS	29
APPENDIX A - DESCRIPTION OF CRITICAL EXPERIMENTS	31
A.1 ORNL Experiments on SM-1 Mockup	31
A.1.1 Scope of Experiments	31
A.1.2 Fuel Element Description	31

TABLE OF CONTENTS (CONT'D)

	<u>Page</u>
A. 2 Alco Experiments on SM-1 ZPE Core I	33
A. 2.1 Scope of Experiments	33
A. 2.2 Fuel Element Description	33
A. 3 Alco Experiments on SM-1A Core I and PM-2A Core I	34
A. 4 Alco Experiments on SM-2	35
A. 4.1 Scope of Experiments	35
A. 4.2 Fuel Element Description	35
A. 5 Martin Company Experiments on ERDL Core	36
A. 5.1 Scope of Experiments	36
A. 5.2 Fuel Element Description	36
A. 6 Martin Company Experiments on PM-1	37
A. 6.1 Scope of Experiments	37
A. 6.2 Fuel Element Description	37
A. 7 Martin Company Experiments on MPR	38
A. 7.1 Scope of Experiments	38
A. 7.2 Fuel Element Description	38
A. 8 Martin Company Experiments on ANPP Cores	39
A. 8.1 Scope of Experiments	39
A. 8.2 Fuel Element Description	39

TABLE OF CONTENTS (CONT'D)

	<u>Page</u>
A.9 ORNL and Phillips Petroleum Experiments on BSR-II	40
A.9.1 Scope of Experiments	40
A.9.2 Fuel Element Description	40
A.10 Phillips Petroleum Experiments on SPERT III	41
A.10.1 Scope of Experiments	41
A.10.2 Fuel Element Description	41
A.11 ANL Experiments on GE. Type Elements	42
A.11.1 Scope of Experiments	42
A.11.2 Fuel Element Description	42
APPENDIX B - TRANSIENT EXPERIMENTS	43
B.1 SM-1 Elements in SPERT I	43
B.2 BSR-II	43
B.3 SPERT III	43
APPENDIX C - REDUCTION OF EXPERIMENTAL DATA	45
REFERENCES	51

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
1	Comparison of 7 X 7 SM-2 Experimental Critical Masses With Predicted Values Using Linear Correlation Model.	6
2	Comparison of Experimental Critical Masses With Predicted Values Using Linear Correlation Model for All 7 X 7 APPR- Type Arrays.	7
3	Fast and Thermal Group Constants of Cores Analyzed by Rand Correlation.	21
4	Physical Geometrical Properties of Critical Cores.	46
5	Summary of Scope of Experiments Performed on Light Water Moderated, Fully Enriched Uranium, Stainless Steel Cores.	49

LIST OF FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
1	Variation of ϕ_{th}/ϕ_{TOT} with $\xi \Sigma_s/\Sigma_a$	24
2	Variation of Buckling Parameter with $\xi \Sigma_s/\Sigma_a$	27

1.0 SUMMARY

The successful application of the semi-empirical relationships of buckling and critical mass in homogeneous cores prompted the present investigation of the use of similar quantities in heterogeneous cores.

A large number of critical experiments have been analyzed, and correlations made. A relationship for the critical mass, involving the stainless steel metal fraction and the boron content of the core, has been developed. A relationship for the buckling has also been found to be dependent on the quantity $\xi \Sigma_s / \Sigma_a$. The range of validity for the buckling lies between 0.007 to 0.023 cm⁻² and covers cores with plate-type elements of highly enriched uranium with stainless steel and light water moderator.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

2.0 INTRODUCTION

Although many experiments have been performed in the field of reactor engineering, little effort has been directed toward establishing universal correlations of this data. The main exception to this is in the realm of reactor safety where a number of highly successful attempts have been made. (1), (2), (3) These attempts were directed at obtaining correlations of such parameters as minimum critical mass and critical bucklings of homogeneous fissionable materials as functions of composition and geometry.

These homogeneous correlations while very useful in establishing conservative safety criteria, are of little value as engineering tools for the design of a power reactor. The success realized in developing these homogeneous correlations, however, casts a favorable light on the possibility of establishing similar correlations of reactor parameters for the heterogeneous power reactors. Such correlations would be extremely useful tools for nuclear engineering applications.

While the ultimate goal in establishing this type of correlation is to develop universal relationships which can be applied to all reactors, the present effort has been restricted to light water moderated, fully enriched uranium, stainless steel reactors of the type being developed in the Army reactor program. Over 70 critical mass experiments have been performed on this type of reactor and it is from these experiments that the correlations are being developed.

The work is divided into two separate categories, collection and reduction of experimental data, and development of a correlation model. A description of the experiments and the techniques used to reduce the experimental data is given in the appendices. The main body of the report describes some of the correlation models investigated and the results of applying these models to the subject type cores. This work was performed under Subtask 2.2. of the Program for Engineering Support and Development of Army PWR Power Plants. (4)

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

3.0 PREVIOUS CORRELATION ATTEMPTS

In the SM-2 project, work was performed to develop a critical mass correlation which could be used to predict the critical masses of 7 x 7 arrays of SM-2 type elements ($B^2 \approx .007$) with varied boron and stainless steel content.⁽⁵⁾ This effort was based on empirical relationships which had been generated from experimental data on other 7 x 7 arrays of APPR type fuel elements. These relationships are discussed in the following subsections.

3.1 EXPERIMENTS ON SM-1 AND SM-1 MOCKUPS

A series of experiments were performed on mockups of the SM-1 (see Appendix A) by ORNL.⁽⁶⁾ Several critical 7 x 7 arrays were constructed with several different metal-to-water ratios and boron loadings. For constant metal-to-water ratios, critical mass was found to vary linearly with boron loading. Several critical cores containing no boron were constructed and critical mass was found to vary linearly with metal fraction.

This linear variation of critical mass with metal fraction and boron loading found in the ORNL experiments was also exhibited in the zero power experiments on the SM-1⁽⁷⁾ and to some extent in the ERDL mockup experiments⁽⁸⁾ on the SM-1.

The linear variation of critical mass with boron loading exhibited by the ORNL experiments and the SM-1 zero power data has been summarized in an analysis performed at ORNL.⁽⁹⁾

3.2 CORRELATION OF 7 x 7 ARRAYS OF APPR- TYPE ELEMENTS

In the SM-2 program, an attempt was made to predict critical mass conditions of SM-2 mockups using the linear relationships shown in the earlier experiments on the SM-1. This data suggests that critical mass may be expressed as:⁽⁴⁾

$$\text{critical mass} = \alpha (\text{metal fraction}) + \beta (\text{grams of B-10}) + m_0 \quad (1)$$

where m_0 , α , and β are constants. These constants were evaluated using the SM-1 data and the resulting equation was used to predict the critical mass of several SM-2 arrays. The agreement with the experimental critical masses was found to be very good except for the core containing no boron. (See Table 1)

The success in predicting the SM-2 critical masses suggested more widespread application of the model for correlating critical mass data. The model, as developed, was based on cores of the same geometrical size. For further application, it must therefore be restricted to cores with approximately the same buckling. As shown on Table 4 (rear of Appendices), experiments have been performed on 20 cores of the APPR type with bucklings of $\sim 0.007 \text{ cm}^{-2}$. Equation (1) was applied to all of these cores and the predicted critical masses are compared with the experimentally determined values in Table 2. The agreement is seen to be good for the ORNL mockups of the SM-1 and for the SM-2 cores. For the ERDL cores and for the SM-1 cores, much poorer results were obtained.

TABLE 1

COMPARISON OF 7 x 7 SM-2 EXPERIMENTAL CRITICAL
MASSES WITH PREDICTED VALUES USING LINEAR COR-
RELATION MODEL

<u>Material Content</u>		<u>Critical Mass</u>	
<u>% Metal</u>	<u>gms B-10</u>	<u>Measured</u>	<u>Calculated</u>
6.58	0	5.30	6.02
8.8	12.3	8.72	9.2
18.1	62.3	22.71	23.0
16.9	58.2	21.40	22.1
7.8	3.52	6.51	7.0

TABLE 2

COMPARISON OF EXPERIMENTAL CRITICAL MASSES WITH
PREDICTED VALUES USING LINEAR CORRELATION MODEL
FOR ALL 7 X 7 APPR- TYPE ARRAYS

$$\text{Critical Mass} = 2.70 + 39.5 (\text{metal fraction}) + 0.065 (\text{gms of B-10})$$

Core	Metal Fraction %	B-10 Loading gm	Calculated Critical Mass	Experim'l Critical Mass
SM-2 Flex. Zero Power Exper. by Alco	6.58	0	5.30	6.02
SM-1 Mockup by ORNL	19.7	0	10.48	10.35
SM-1 Mockup by ORNL	18.2	0	9.88	9.9
SM-1 Mockup by ORNL	9.5	0	6.45	6.6
SM-1 Mockup by Martin (ERDL)	18.1	0	9.85	13.25*
SM-1 Mockup by ORNL	19.7	4.15	11.34	11.32
SM-1 Mockup by ORNL	19.7	6.39	11.80	11.76
SM-1 Mockup by ORNL	19.7	15.56	13.69	13.60
SM-1 Mockup by ORNL	19.7	23.33	15.30	15.18
SM-1 Mockup by ORNL	19.7	31.11	16.90	16.67
SM-1 Mockup by ORNL	19.7	38.89	18.51	18.52
SM-1 Mockup by ORNL	19.7	46.67	20.12	20.10
SM-1 Zero Power Exper. by Alco	18.5	27.80	17.82	22.48
SM-1 Zero Power Exper. by Alco	36.1	19.52	20.98	22.48
SM-2 Flex. Zero Power Exper. by Alco	8.8	12.28	8.72	9.2
SM-2 Flex. Zero Power Exper. by Alco	18.1	62.3	22.71	23.0
SM-2 Flex. Zero Power Exper. by Alco	16.9	58.2	21.40	22.1
SM-2 Flex. Zero Power Exper. by Alco	7.8	3.52	6.51	7.0

* Part of discrepancy was due to presence of voids

In an attempt to expand the versatility of the model, a series of linear equations were set up to incorporate the effect of other parameters which affect the critical mass including different geometries and core sizes. Some of these equations used are given below. In Eq. (2), the Greek symbols represent constants, the N_i 's are atomic concentrations of various materials and B^2 is the critical buckling.

$$\begin{aligned} \text{critical mass} &= m_0 + \alpha(MF) + \beta(\text{grams of B-10}) + \gamma\left(\frac{N_h}{N_u}\right) \\ \text{critical mass} &= m_0 + \alpha(MF) + \beta(\text{grams of B-10}) + \gamma\left(\frac{N_h}{N_u}\right) + \delta(B^2) \quad (2) \\ \text{critical mass} &= m_0 + \alpha(MF) + \beta(\text{grams of B-10}) + \gamma\left(\frac{N_h}{N_u}\right) + \delta(B^2) + \lambda(N_u) \end{aligned}$$

In attempting to fit the critical mass values, data was taken from the cores in Table 4. To simplify the model, several fits were attempted using cores containing no boron. The results obtained using Eq. (2) were quite poor and it appeared that the linear model was not a satisfactory method for general application.

3.3 DISCUSSION OF LINEAR CORRELATION MODEL

While the initial results of applying the linear correlation model were quite successful, attempts to improve and expand its versatility failed. Application of the model to the SM-1 mockup in the ERDL experiments and to the zero power experiments on the SM-1 were particularly unsuccessful. Since the only difference between the ERDL cores and the ORNL mockups of the SM-1 was in the degree of heterogeneity, ⁽⁹⁾ it appeared that some measure of heterogeneity effects would have to be incorporated if the generality of the model were to be improved.

The inclusion of heterogeneity effects required a re-examination of the results of the SM-2 calculations. In the SM-2 experiments, the fuel is dispersed in a stainless steel matrix forming the fuel plates, while the boron is incorporated in Mylar tape and is mechanically attached to the fuel plates. (See Appendix A.) In the ORNL experiments, the elements were mocked up with a sandwich type construction where the fuel was inserted in the form of uranium foils. Boron plates were fabricated by dispersing B_4C in an iron matrix, clad with stainless steel. There is an obvious difference in the degree of heterogeneity between the two, yet the SM-1 data was used to obtain the constants in Eq. (1) which in turn was used to predict the SM-2 critical masses. This implies that the success in predicting the SM-2 critical masses may have been coincidental.

Another indication of the problems associated with this type of model may be seen from the ORNL analysis.⁽⁹⁾ ORNL obtained plots of critical mass vs. boron loading at constant metal-to-water ratios. A linear dependence was observed, but it was necessary to plot the foil type cores on a separate curve from the fuel in matrix type cores. Both curves have about the same slope but with different intercepts.

One other problem associated with the linear model should be mentioned. The constants α and β in Eq. (1) may vary from system to system. Use of the equation in its present form implies that all cores of this size would have the same coefficients. Experiment does not support this hypothesis.

Failure of the linear correlation model to lend itself to more general applications suggested that better founded techniques should be used. Accordingly, the feasibility of using some of the techniques employed in the correlations of homogeneous reactor parameters was investigated. A correlation developed by Pinkel and Young⁽¹⁰⁾ seemed particularly well suited for modification for the present applications. A discussion of this model and the results of applying it to the subject type cores is contained in Section 4.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

4.0 CRITICAL MASS CORRELATION FOR BARE HOMOGENEOUS REACTOR

The basic theory involved in the correlation developed here differs only slightly from a correlation developed by Pinkel and Young. (10) Their work was restricted to cores containing homogeneous mixtures of U-235 and various moderators. A method was suggested for incorporating the effects of secondary materials in the correlation. However, the nature of the correlating parameter which they developed demands that these secondary materials be characterized by relatively small neutron absorption cross-sections with $\frac{1}{\sqrt{E}}$ energy dependence. The following development differs from the work of Pinkel and Young in the selection of a correlating parameter which does not restrict the validity of the correlation to low absorption secondary materials.

The standard multi-group diffusion equation for the i th group in an m group breakdown is:

$$\nabla \cdot D_i \nabla \phi_i + f_i \nu \sum_{j=1}^m \Sigma_{fj} \phi_j + \sum_{j=1}^{i-1} \Sigma_{sji} \phi_j - \Sigma_{ai} \phi_i - \Sigma_{si} \phi_i = 0 \quad (3)$$

where

- D_i is the diffusion coefficient in the i th group
- ϕ_i the neutron flux of group i
- f_i the fraction of neutrons born in the i th group due to fission
- ν the average number of neutrons produced per fission average
- Σ_{fj} the average macroscopic fission cross section of U-235 in the j th energy group
- Σ_{sji} the macroscopic scattering cross section resulting in a neutron slowing from the j th group to the i th group
- Σ_{ai} the average macroscopic absorption cross section in the i th group
- Σ_{si} the macroscopic slowing down cross section for slowing neutrons from the i th group to a lower energy group.

and the m th group is the lowest energy, or thermal, group.

The first term in Eq. (3) represents the net flow of neutrons having energy E from a unit volume element. The second term represents the generation of neutrons due to fission in the i th energy group. The third term represents the neutrons scattered into the i th energy group from higher energy groups. The fourth term represents the loss of neutrons from the i th group due to capture and the last term in Eq. (3) represents the neutrons which are lost from the i th group due to slowing down into lower energy groups. If the m groups are now summed over the complete energy range, we obtain:

$$\sum_{i=1}^m \nabla \cdot D_i \nabla \phi_i + \sum_{i=1}^m f_i \nu \sum_{j=1}^m \Sigma_{fj} \phi_j + \sum_{i=1}^m \sum_{j=1}^{i-1} \Sigma_{sji} \phi_j - \sum_{i=1}^m \Sigma_{ai} \phi_i - \sum_{i=1}^{m-1} \Sigma_{si} \phi_i = 0 \quad (4)$$

The term $\sum_{i=1}^m \sum_{j=1}^{i-1} \Sigma_{sji} \phi_i$ represents the total number of neutrons slowed into all energy groups and the term $\sum_{i=1}^{m-1} \Sigma_{si} \phi_i$ represents the total number of neutrons scattered out of all energy groups. These two slowing down terms are obviously equal and being of opposite sign, cancel. By definition

$\sum_{i=1}^m f_i \nu = \nu$. Equation (4) therefore reduces to:

$$\sum_{i=1}^m \nabla \cdot D_i \nabla \phi_i + \nu \sum_{i=1}^m \Sigma_{fi} \phi_i - \sum_{i=1}^m \Sigma_{ai} \phi_i = 0 \quad (5)$$

Multiplying and dividing by $\phi_{TOT.} = \sum_{i=1}^m \phi_i$ and rearranging terms yields Eq. 6.

$$\sum_{i=1}^m \nabla \cdot D_i \nabla \left(\frac{\phi_i}{\phi_{TOT.}} \phi_{TOT.} \right) + \sum_{i=1}^m \left(\nu \Sigma_{fi} - \Sigma_{ai} \right) \frac{\phi_i}{\phi_{TOT.}} \phi_{TOT.} = 0 \quad (6)$$

If the analysis is now restricted to a bare homogeneous system, it can be assumed that D_i and $\frac{\phi_i}{\phi_{TOT.}}$ are independent of the space coordinates, hence:

$$\sum_{i=1}^m D_i \frac{\phi_i}{\phi_{TOT.}} \nabla^2 \phi_{TOT.} + \sum_{i=1}^m \left(\nu \Sigma_{fi} - \Sigma_{ai} \right) \frac{\phi_i}{\phi_{TOT.}} \phi_{TOT.} = 0 \quad (7)$$

The material buckling, B, is defined as:

$$B^2 = \frac{\sum_{i=1}^{i=m} \left(\nu \Sigma_{f_i} - \Sigma_{a_i} \right) \frac{\phi_i}{\phi_T}}{\sum_{i=1}^{i=m} D_i \frac{\phi_i}{\phi_T}}, \quad (8)$$

so that Eq. (7) reduces to the form:

$$\nabla^2 \phi_T + B^2 \phi_T = 0 \quad (9)$$

We now define our parameters as effective flux weighted values by letting

$$\begin{aligned} \nu \Sigma_f &= \sum_{i=1}^{i=m} \left(\nu \Sigma_{f_i} \frac{\phi_i}{\phi_T} \right) \\ \Sigma_a &= \sum_{i=1}^{i=m} \left(\Sigma_{a_i} \frac{\phi_i}{\phi_T} \right) \\ \lambda_t &= 3 \sum_{i=1}^{i=m} \left(D_{i_a} \frac{\phi_i}{\phi_T} \right) \end{aligned} \quad (10)$$

Our expression for B^2 then becomes

$$B^2 = \frac{3}{\lambda_t} \left(\nu \Sigma_f - \Sigma_a \right) \quad (11)$$

Equation (11) may be written in expanded form as:

$$1/3 \lambda_t B^2 = N_u \sigma_{au} \left(\frac{\nu \Sigma_f}{N_u \sigma_{au}} - 1 - \frac{N_h \sigma_{ah}}{N_u \sigma_{au}} - \frac{N_b \sigma_{ab}}{N_u \sigma_{au}} - \frac{N_{ss} \sigma_{ass}}{N_u \sigma_{au}} \right) \quad (12)$$

where,

$$\Sigma_a = N_u \sigma_{au} + N_h \sigma_{ah} + N_b \sigma_{ab} + N_{ss} \sigma_{ass}$$

$$\Sigma_f = N_u \sigma_{fu}$$

$$B^2 = B_g^2 = B_m^2 \text{ at criticality}$$

$\sigma_{a's}$ = effective flux weighted microscopic cross sections

but,

$$\frac{1}{N_u} = \frac{M_u}{\rho_u v A} = \frac{21.12}{v}$$

where

M_u = molecular weight of U-235

ρ_u = density of pure U-235

v = volume fraction of U-235

A = Avogadro's number

Hence,

$$\frac{7.04 \lambda_t B^2}{v} = (\nu \sigma_f - \sigma_{au}) - \sigma_{ah} \frac{N_h}{N_u} - \sigma_{ab} \frac{N_b}{N_u} - \sigma_{ass} \frac{N_{ss}}{N_u} \quad (13)$$

The first term on the right is dependent only on the flux spectra. Hence, $(\nu \sigma_f - \sigma_{au}) = F_1(\phi)$ where $F_1(\phi)$ is some function of the flux spectrum. We may therefore write:

$$\frac{7.04 \lambda_t B^2}{v} = F_1(\phi) - \sigma_{ah} \frac{N_h}{N_u} - \sigma_{ab} \frac{N_b}{N_u} - \sigma_{ass} \frac{N_{ss}}{N_u} \quad (14)$$

The terms

$$\sigma_{ah} \frac{N_h}{N_u}, \sigma_{ab} \frac{N_b}{N_u}, \sigma_{ass} \frac{N_{ss}}{N_u}$$

may be written as:

$$\sigma_{aho} \frac{\sigma_{ah}}{\sigma_{aho}} \frac{N_h}{N_u}, \sigma_{abo} \frac{\sigma_{ab}}{\sigma_{abo}} \frac{N_b}{N_u}, \sigma_{asso} \frac{\sigma_{ass}}{\sigma_{asso}} \frac{N_{ss}}{N_u}$$

where σ_{aho} , σ_{abo} , σ_{asso} are microscopic cross sections evaluated at some reference energy. Equation (15) may now be written as:

$$\begin{aligned} \frac{7.04 \lambda_t B^2}{v} = F_1(\phi) - \sigma_{aho} \frac{\sigma_{ah}}{\sigma_{aho}} \frac{N_h}{N_u} - \sigma_{abo} \frac{\sigma_{ab}}{\sigma_{abo}} \frac{N_b}{N_u} \\ - \sigma_{asso} \frac{\sigma_{ass}}{\sigma_{asso}} \frac{N_{ss}}{N_u} \end{aligned} \quad (15)$$

The cross sections of water and boron exhibit an energy dependence of the form $\frac{1}{\sqrt{E}}$.

A $\frac{1}{\sqrt{E}}$ dependence has been assumed for stainless steel cross section.

Incorporating this assumption, the cross section ratios,

$\frac{\sigma_{ah}}{\sigma_{aho}}, \frac{\sigma_{ass}}{\sigma_{asso}}, \frac{\sigma_{ab}}{\sigma_{abo}}$ are not only functions of the flux spectrum, but depend

on the same function, which is defined as $F_2(\phi)$. Equation (15) may now be written as:

$$\frac{7.04 \lambda_t B^2}{v} = F_1(\phi) - \left(\sigma_{aho} \frac{N_h}{N_u} + \sigma_{abo} \frac{N_b}{N_u} + \sigma_{asso} \frac{N_{ss}}{N_u} \right) F_2(\phi) \quad (16)$$

This can be rewritten as:

$$\frac{7.04 \lambda_t B^2}{v} = F_1(\phi) - \sigma_{aho} \frac{N_h}{N_u} \left(1 + \frac{\sigma_{abo}}{\sigma_{aho}} \frac{N_b}{N_h} + \frac{\sigma_{asso}}{\sigma_{aho}} \frac{N_{ss}}{N_h} \right) F_2(\phi) \quad (17)$$

or, letting

$$\epsilon = \left(1 + \frac{\sigma_{abo}}{\sigma_{aho}} \frac{N_b}{N_h} + \frac{\sigma_{asso}}{\sigma_{aho}} \frac{N_{ss}}{N_h} \right)$$

$$\frac{7.04 \lambda_t B^2}{v} = F_1(\phi) - \sigma_{aho} \frac{N_h}{N_u} \epsilon F_2(\phi) \quad (18)$$

The equation has therefore been reduced to two functions dependent only on the flux spectrum.

The parameter $\xi \Sigma_s(E)/\Sigma_a(E)$ may be regarded as a measure of the mean neutron energy degradation per capture. It is not unreasonable to assume that to a first order approximation, systems with a given set of materials having the same value of this parameter would also have the same flux spectrum. In the analysis by Pinkel and Young,⁽¹⁰⁾ a similar assumption was made except their parameter was expressed as

$$\epsilon y_m \frac{N_h}{N_u}, \text{ where for our conditions,}$$

$$y_m = \frac{(\xi \sigma_s)_H}{(\xi \sigma_s)_{D_2O}}$$

Their parameter differs from $\xi \Sigma_s / \Sigma_a$ in that σ_{au} was assumed constant and factored out. In Pinkel and Young's correlation, the cores investigated contained only fuel combined with various moderators. The present study considers only light water moderated systems but as second and third variables includes various quantities of stainless steel and boron which also affect the flux spectrum.

Since our objective is to establish a parameter which will correlate variables dependent primarily on the flux spectrum, the first step must, of

necessity, be to establish a parameter which will correlate flux spectra in the subject type cores. Correlation of the flux spectrum should then lead to the evaluation of the functions $F_1(\phi)$ and $F_2(\phi)$. This is done in Sections 6.0 and 7.0.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

5.0 CALCULATIONAL TECHNIQUES

The correlation model as developed in Section 4.0 is valid only for bare homogeneous systems. The techniques employed in modifying the heterogeneous reflected reactors under consideration to equivalent bare homogeneous systems are discussed below.

5.1 EQUIVALENT HOMOGENEOUS SYSTEMS

There are two types of heterogeneities present in many of the critical cores being investigated. In addition to the heterogeneities in an element cell, there also exist differences in composition in different cells due to the presence of both stationary and control rod follower fuel elements.

5.1.1 Cell Calculations

The modification of a heterogeneous element cell to an equivalent homogeneous cell is a simple process as far as obtaining the nuclear properties for the fast neutron group is concerned. Here self-shielding is essentially negligible because there are no appreciable resonances in the fully enriched lattices under consideration, based on element number densities homogenized over the entire cell. The nuclear constants were obtained with the aid of the MUFT III Code for the IBM-650.(11)

The self-shielding due to heterogeneities is a severe problem in determining the thermal group nuclear constants in the subject cores. This is treated by performing a detailed calculation of the flux distribution in an element cell. Effective nuclear constants are then defined which will yield the same reaction rate in a homogeneous cell as in the heterogeneous cell. The effective cross sections are defined as:

$$\sum_i^{eff} = \frac{\int_{cell} \Sigma_i(x,y) \phi(x,y) dx dy}{\int_{cell} \phi(x,y) dx dy} \quad (20)$$

Transport theory is used to calculate the detailed flux distribution in the heterogeneous cell. For the present application, effective nuclear parameters were obtained with the aid of the "APWRC - CELCOR" code(12) for the IBM-7090, and the "Two Dimensional P₃ Calculation for APPR - Type Fixed Fuel Elements"(13) Code for the IBM-650. The nuclear constants for the fast and thermal groups for the cores analyzed are given in Table 3.

5.1.2 Core Calculations

Having established the effective homogeneous nuclear constants for the individual elements, effective constants were then calculated for the entire core. This was necessary because of the differences in composition between the stationary elements and the control rod elements. Using the effective constants for the individual elements, a two group, several region calculation was performed and effective core constants were obtained using Eq. (20), where the integration limits now included the entire core.

5.2 EQUIVALENT BARE CORES

The reflector savings for most of the cores being considered in this study were determined from experimental flux traverses. These are discussed in Appendix C. This data is not available for all of the cores however, and the geometries of the others is such that accurate bucklings can not be determined from the flux traverses. Reflector savings were evaluated for the latter cores by means of an analytical solution to the two group, two region problem in cylindrical geometry.

TABLE 3

FAST AND THERMAL GROUP CONSTANTS OF CORES ANALYZED IN RAND CORRELATION

	Core Description	p	$\nu \sum_f^f$	$\sum_a^f$	$\xi \sum_s^f$	D_f	τ	$\sum_a^{th}$	$\nu \sum_f^{th}$	D_{th}	L^2
BSR-2	20 full elements and 1-1/4 element	.79555	.010076	.008153	.031544	1.27642	32.728	.19419	.31760	.18110	.9325
SM-2	7 x 7 corners remove	.922570	.003898	.003444	.041212	1.2664	28.37	.058217	.07554	.186560	3.2044
PM-1	Minimum size critical core	.733876	.015442	.010466	.028860	1.25499	31.9127	.279871	.501316	.164367	.58730
ANPP	3 x 3 core No. 450	.589774	.021840	.015315	.022018	1.18878	31.8428	.31528	.549346	.186071	.59017
ANPP	3 x 2-7/18 core No. 350	.627865	.022178	.014839	.025036	1.20219	30.1486	.312611	.555222	.175057	.56000
ANPP	3 x 2-14/18 core No. 400	.600939	.022133	.015243	.022954	1.19435	31.2685	.320931	.564059	.179976	.56080
PM-1	Core with 19, 0.27 wt % boron tubes	.73556	.014128	.010811	.030072	1.21511	29.7212	.288028	.45185	.16575	.5775
PM-1	Core with 13, 0.27 wt % boron tubes	.744858	.013728	.010867	.031723	1.15223	27.054	.285176	.462197	.16528	.580
ANPP	5 x 5 core No. 454	.545651	.021067	.017083	.020515	1.9397	31.7564	.377580	.520997	.181448	.48056
ANPP	4 x 4-8/18 core No. 453	.559619	.021402	.016633	.021137	1.19381	31.6075	.363155	.533990	.180881	.49807
ANPP	4 x 3-12/17 cores No. 452	.570578	.021412	.016125	.021426	1.19342	31.7803	.350596	.543256	.181627	.51806
ANPP	5 x 4-1/18 core No. 354	.585288	.020903	.016443	.023206	1.20678	30.4364	.380611	.537306	.175212	.46033
ANPP	4 x 3-12/18 core No. 353	.596203	.021606	.016176	.023884	1.20631	30.1121	.367568	.535240	.174042	.47348
ANPP	4 x 3-3/18 core No. 352	.604976	.021687	.015728	.024088	1.20428	30.2462	.354815	.563439	.174520	.49213
ANPP	4 x 4 core No. 403	.573642	.021662	.016551	.022268	1.19752	30.8490	.364133	.540584	.178438	.49004
SM-2	Min. size- 3x3 18 plates/ el and 16 plates/C.R. el	.65050	.017773	.012569	.023655	1.26357	35.169	.327940	.56700	.17212	.472

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

6.0 CORRELATION OF FLUX SPECTRA

As stated in Section 4.0, the success of the correlation depends primarily on the ability of the parameter $\frac{\xi \Sigma_s}{\Sigma_a}$ to correlate flux spectra. The ratio of the average thermal flux to the total flux is a convenient measure of the flux spectra. This ratio may be obtained by solving the two group equations to yield:

$$\frac{\phi_f}{\phi_{th}} = \frac{k_{eff} \Sigma_a^{th}}{pD_f/\tau} \left[\frac{1}{1 + \tau B^2} \right] \quad (21)$$

The relative thermal flux is then obtained by normalizing the total flux, $\phi_{th} + \phi_f$, to 1.0. The nuclear parameters used in Eq. (20) for the various cores investigated were taken from Table 3. The variation $\frac{\phi_{th}}{\phi_{TOT}}$ for these cores is shown in Fig. 1.

From Fig. 1, it is seen that $\frac{\phi_{th}}{\phi_{TOT}}$ exhibits a regular variation with $\frac{\xi \Sigma_s}{\Sigma_a}$. In the range investigated, the variation of $\frac{\phi_{th}}{\phi_{TOT}}$ is nearly linear and a least squares fit of the available data points results in Eq. (22). The root mean square deviation of the least squares is 0.031.

$$\frac{\phi_{th}}{\phi_{TOT}} = -0.0237 + 0.0285 \left(\frac{\xi \Sigma_s}{\Sigma_a} \right) \quad (22)$$

It may be concluded that $\frac{\xi \Sigma_s}{\Sigma_a}$ represents a satisfactory parameter for the correlation of flux spectra.

Figure 1. Variation of ϕ_{th}/ϕ_{TOT} with $\xi \Sigma_s/\Sigma_a$

7.0 CORRELATION OF CRITICAL BUCKLING

In Section 4 it was shown that the critical buckling may be expressed as:

$$\frac{7.04 \lambda_t B^2}{v} = F_1(\phi) - \frac{N_h}{N_u} \sigma_{amo} \in F_2(\phi) \quad (23)$$

The functions $F_1(\phi)$ and $F_2(\phi)$ are dependent only on the flux spectrum.

Since the flux spectrum may be correlated by the parameter $\frac{\xi \Sigma_s}{\Sigma_a}$, $F_1(\phi)$

and $F_2(\phi)$ should also be primarily dependent on $\frac{\xi \Sigma_s}{\Sigma_a}$. We may therefore write Eq. (23) as:

$$\frac{7.04 \lambda_t B^2}{v} = F_1\left(\frac{\xi \Sigma_s}{\Sigma_a}\right) - \frac{N_m}{N_u} \sigma_{amo} F_2\left(\frac{\xi \Sigma_s}{\Sigma_a}\right) \quad (24)$$

From Pinkel and Young's data, (10) it can be shown that $F_2\left(\frac{\xi \Sigma_s}{\Sigma_a}\right)$ is vanishingly small for cores in which $\frac{\phi_{th}}{\phi_{TOT}}$ is on the order of 10% or less.

From Fig. 1, it is seen that $\frac{\phi_{th}}{\phi_{TOT}}$ is less than .16 for all the cores investigated. For purposes of this analysis, it will be assumed that Eq. (24) may be written in an approximate manner as:

$$\frac{7.04 \lambda_t B^2}{v} \cong F_1\left(\frac{\xi \Sigma_s}{\Sigma_a}\right) \quad (25)$$

The variation of the buckling parameter $\frac{7.04 \lambda_t B^2}{v}$ is shown in Fig. 2 as a

function of $\frac{\xi \Sigma_s}{\Sigma_a}$. Using the method of least squares, a linear fit of the

available data points has been made. For the cores investigated, an expression of the form

$$\frac{7.04 \lambda_t B^2}{v} = -18.89 + 10.35 \left(\frac{\xi \Sigma_s}{\Sigma_a}\right) \quad (26)$$

fits the data with a root mean square deviation of 0.062.

7.1 DISCUSSION OF BUCKLING CORRELATION

Reference to Fig. 2 indicates that Eq. (26) represents a reasonable accurate fit of the available experiments. This equation can, therefore, be used to calculate the critical mass of U^{235} given the core buckling and inventory of stainless steel and Boron-10.

Figure 2. Variation of Buckling Parameter with $\xi \Sigma_s / \Sigma_a$

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

8.0 CONCLUSIONS AND RECOMMENDATIONS

1. An equation of the form -

$$\text{Critical Mass} = C_1 + C_2 (\text{SS Fraction}) + C_3 (\text{gm B-10})$$

has been found which yields reasonable agreement with available critical experiment cores of SM-1 size (geometrical buckling = 0.007 cm^{-2}).

2. All of the available experiments have been found to fit an equation of the form $B^2 = C_1 + C_2 \left(\frac{\xi \Sigma_s}{\Sigma_a} \right)$. This equation can then be used to determine the critical mass of U^{235} for cores having $\left(\frac{\xi \Sigma_s}{\Sigma_a} \right)$ in the range 3 to 6.5 and bucklings in the range 0.007 to 0.023 cm^{-2} .

3. The relative accuracy of the two methods of correlating experiments should be critically evaluated.

4. The large body of critical experiment information on light water moderated, fully enriched uranium, stainless steel reactors should be the subject of additional efforts to develop a correlation method, in order to provide for accurate predictions of critical mass of cores of this class.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX A - DESCRIPTION OF CRITICAL EXPERIMENTS

Experiments on light water moderated, fully enriched uranium, stainless steel cores have been performed by Alco Products, The Martin Company, Oak Ridge National Laboratory, Phillips Petroleum, and Argonne National Laboratory. The scope of these experiments is discussed briefly in the following sections. A brief description of the fuel elements used in the experiments is also given. A tabulation of the critical masses determined in the experiments is given in Table 4 (rear of Appendices). The data has been segregated in terms of clean critical cores (no boron) with all control rods fully withdrawn, critical cores containing boron but with all control rods fully withdrawn, and critical cores with one or more control rods fully inserted, all other rods fully withdrawn. A tabulation showing the scope of the experiments is given in Table 5 (rear of Appendices).

A.1 ORNL EXPERIMENTS ON SM-1 MOCKUP

One of the first sets of experiments on the subject cores of interest was performed by ORNL in 1955⁽⁶⁾. These experiments were conducted for purposes of predicting the nuclear characteristics of the SM-1 (formerly APPR-1) using mockup fuel elements.

A.1.1 Scope of Experiments

These experiments included determinations of five clean critical masses at 68°F with different volume fractions of metal. In addition, the critical mass was determined for 10 different systems with various boron loadings but with a constant metal-to-water ratio. Flux and power distribution measurements were made. An estimate of the effects of heterogeneity was determined by the substitution of a homogeneous element. The effect of voids was also measured.

A.1.2 Fuel Element Description⁽⁶⁾

The fuel elements used in these experiments were composed of a varying number of fuel plates up to a maximum of 18. Each individual fuel plate consisted of an enriched (93.2%) uranium foil encased in type 304 stainless steel sheets. The edges of the bottom stainless steel sheet were folded over the top piece completely enclosing the foil, which in turn was glued to both steel sheets. This was done to prevent water from coming in contact with

the foil. The dimensions of the foils were 2.50 x 0.002 x 22.0 in. The over-all dimensions of the fuel plate were 2.723 x 0.025 x 23.0 in.

For the experiments performed with boron in the core, boron plates were fabricated by using B_4C in a sintered iron matrix clad with type 304 stainless steel. The clad-core-clad dimensions were 0.005, 0.015, and 0.005 in. respectively. The over-all dimensions were the same as those of the fuel plates. This allowed boron to be inserted by replacement of a fuel or a steel plate without changing the metal-to-water ratio.

To obtain various metal-to-water ratios with the desired fuel loadings, stainless steel plates (type 304) were made to the same dimensions as the fuel plates, and added as desired.

The fuel, boron, and steel plates were in a fuel box of type 304 stainless steel. Two opposite sides of each box contained 18 grooves, 0.050 x 0.56 in. forming slots which held the plates. The over-all box dimensions were 2.81 x 2.85 x 31 in.

For flexibility in fuel loadings, plates were also made with one half width uranium foils.

A. 2 ALCO EXPERIMENTS ON SM-1 ZPE CORE I*

A. 2.1 Scope of Experiments

In the extensive series of experiments on the SM-1 ZPE Core I, (7, 15) critical mass determinations were made for 5 x 5, 6 x 6, and 7 x 7 arrays which were poisoned by the addition of stainless steel and of boron-stainless steel. In addition, the minimum critical mass of an array of standard SM-1 Core I elements was determined. Thermal and fast flux distributions were obtained for the 5 x 5, 6 x 6, and 7 x 7 arrays.

Material reactivity coefficients were evaluated and the temperature coefficient was determined by displacing water with aluminum strips. The effects of adding a stainless steel reflector were also investigated.

A. 2.2 Fuel Element Description

The SM-1 ZPE Core I utilized two distinct types of fuel elements. (16) These include the stationary fuel elements and the control rod fuel elements. The stationary elements consist of 18 fuel plates held in place by two slotted side plates. The control rod fuel elements are connected to the absorber sections and are pulled into the core as the control rod elements are withdrawn. The control rod elements contain 16 fuel plates and are also held in place by slotted side plates.

The fuel plates in both element designs are sandwich type. The fuel and boron are dispersed in a steel matrix in the form of UO_2 and B_4C respectively.

* SM-1 ZPE Core I was made to the same specifications (16) as SM-1 Core I

A.3 ALCO EXPERIMENTS ON SM-1A CORE I AND PM-2A CORE I

Minimum critical mass determinations were also made on the SM-1A Core I⁽¹⁷⁾ and PM-2A Core I⁽¹⁸⁾. The fuel elements in these cores are very similar to those in the SM-1 ZPE Core I differing only by small changes in composition.

A.4 ALCO EXPERIMENTS ON SM-2 (19), (20)

Flexible and mockup experiments were performed in support of the SM-2 design. Fuel plates consisting of UO_2 dispersed in SS and clad with SS were employed. Boron was added in the form of boron bearing Mylar tape. Various numbers of fuel plates could be added to each cell.

A.4.1 Scope of Experiments

Utilizing the flexibility of the elements, an SM-1 mockup was constructed. This was utilized to obtain U-235 and boron reactivity worths in the SM-1. The measurements were repeated for an SM-2 configuration at 68°F and at a simulated temperature of 510°F. Measurements were made to determine the effect of varying the thickness of the stainless steel reflector in the SM-2. Aluminum strips were utilized to mockup the temperature coefficient. Bare gold, bare uranium, and cadmium covered uranium foils were used to determine flux and power distributions in the 7 x 7 SM-2 array.

Two critical configurations containing no boron and with the control rods completely withdrawn were constructed. Four critical configurations were constructed with the elements containing various amounts of boron and with different metal-to-water ratios.

A.4.2 Fuel Element Description

The mockup SM-2 elements are similar to the SM-1 elements with a few notable exceptions. Fuel plate thickness was increased from 0.030 in. to 0.040 in., fuel loading per plate was increased from 28.62 gm to 46.30 gm.

The poison was added in the form of boron impregnated Mylar tape, thereby allowing flexibility in the poison load.

A. 5 MARTIN COMPANY EXPERIMENTS ON ERDL CORE (8)

The ERDL experiments were designed to provide a means for evaluating the characteristics of the APPR (SM-1) prototype and advanced systems of the APPR family. Some difficulty was experienced because of void formations and heterogeneity effects in these experiments.

A. 5.1 Scope of Experiments

Six critical cores, with all control rods fully withdrawn, were constructed. Three of these contained no boron. Various metal-to-water ratios and boron loadings were used. A full core (45 elements) was also constructed on which flux and power distribution measurements were made. Fuel worth measurements were made as a function of radial position in the 45-element core. The worth of boron was determined in one core position.

A. 5.2 Fuel Element Description

The fuel elements used in the ERDL experiments are of the same type used in the ANPP cores; i. e., the plates resemble a multi-layered sandwich. A typical plate contains the following:

0.010 in. of stainless steel

0.0018 in. of uranium foil

0.005 in. of stainless steel

0.0042 in. of boron steel

0.005 in. of stainless steel

0.0018 in. of uranium foil

0.010 in. of stainless steel

When it is desired to construct a core containing no boron, the boron steel section is replaced by stainless steel. Half-width fuel plates were also manufactured and used extensively.

The fuel elements were similar to the elements used in the ORNL experiments on SM-1. A varying number of fuel bundles could be used, up to a maximum of 18, in each element.

A.6 MARTIN COMPANY EXPERIMENTS ON PM-1 (21)

A relatively extensive testing program was carried out to evaluate the nuclear properties of the PM-1 and to evaluate the calculation models used in predicting the core operating characteristics.

A.6.1 Scope of Experiments

The minimum critical mass with all rods fully withdrawn was determined for a core containing no boron. Four other critical cores with rods fully removed were constructed. The latter cores contained various amounts of discrete boron poison and stainless steel. Flux and power distributions were measured on these as well as the other cores which were constructed.

Control rod worth was evaluated as a function of the number and location of boron rods. Temperature coefficients were measured and epicadmium europium rod worths were evaluated.

A.6.2 Fuel Element Description

The PM-1 elements are tubular with a 0.500 in. OD and a 0.416 in. ID. The active length is 30 in. The uranium dioxide is contained in a sintered stainless steel compact. Fuel loading per tube is 39.4 gm U-235. Boron poison is incorporated in stainless steel rods with a 0.500 in. OD.

A.7 MARTIN COMPANY EXPERIMENTS ON MPR⁽²²⁾

The forerunner to the PM-1 was the MPR, a core of tubular elements designed and tested by the Martin Company.

A.7.1 Scope of Experiments

A cold clean critical mass with all control rods removed was determined. Two larger cores were also constructed in which flux distributions were measured.

A.7.2 Fuel Element Description

The tubular fuel elements in the MPR have an outside diameter of 0.375 in. and an inside diameter of 0.315 in. Fuel in the form of UO_2 and poison in the form of B_4C are contained in a stainless steel cormat. Each tube contained 11.96 gm U-235 and 0.09 gm of natural boron.

A. 8 MARTIN COMPANY EXPERIMENTS ON ANPP CORES (23, 24, 25, 26, 27, 28, 29, 30, 31)

The cores studies under the ANPP program are unique in that the sole purpose of the experiments was to provide data which could be used to verify analytical models being developed under the program. For this reason, the cores were quite symmetric and contain few perturbations such as those due to control rod elements, etc., that are found in the other cores.

A. 8.1 Scope of Experiments

Eighteen critical mass determinations were made, eight of which were on cores containing no boron. Boron loading and metal-to-water ratio were the varying parameters in the other cores. Material worth measurements were made for all the cores. Experimental bucklings were obtained by activating thorium, cadmium covered gold, and dysprosium foils, and fitting the data with a cosine curve.

A. 8.2 Fuel Element Description

The fuel plates used in the ANPP cores are of the "club sandwich" type. A typical plate contains the following: 0.017 in. of stainless steel, 0.0018 in. of uranium foil, 0.010 in. of stainless steel, 0.0058 in. of boron steel, 0.010 in. of stainless steel, 0.0018 in. of uranium foil and 0.017 in. of stainless steel. For cores containing no boron, the boron steel was replaced by stainless. The fuel plates were grouped into fuel elements in the same manner as in the ERDL experiments⁽⁸⁾ and the ORNL experiments on the SM-1.⁽⁶⁾ The maximum number of plates in any element is 18.

A.9 ORNL AND PHILLIPS PETROLEUM EXPERIMENTS ON BSR-II

Prior to insertion in the BSR facility, a series of experiments was performed on the BSR-II core to determine some of the static and dynamic characteristics of the system.

A.9.1 Scope of Experiments

Fourteen core configurations were investigated at ORNL. (32) Of these, eight were within 50 cents of criticality with all control rods fully withdrawn. Flux distribution measurements were made on the core at the SPERT I facility by Phillips Petroleum. (33)

A.9.2 Fuel Element Description

The BSR-II elements are box type with both stationary and control rod elements containing 20 fuel plates. The standard element, with full fuel plates, contains 290 gm of U-235. Some elements with one-half and one-quarter of the nominal fuel loading were also used.

The control rod elements differ from the standard element in that the width is reduced to allow space for passage of the poison control plates. Each element is 15 in. long with a 9 in.² square cross section.

The fuel is in a stainless steel cermet, clad with stainless steel.

A.10 PHILLIPS PETROLEUM EXPERIMENTS ON SPERT III

Prior to the startup of the SPERT III reactor, a series of experiments was conducted in the SPERT I facility, using SPERT III type 1S elements.⁽³⁴⁾ Additional experiments ⁽³⁵⁾ were later performed at the SPERT III facility using 1S, 2S, and 3S type elements.

A.10.1 Scope of Experiments

Eleven core configurations were studied, of which five contained one fully inserted control rod. These experiments were of a critical water height nature and only one of them contained less than ~50 cents excess reactivity with the water at the top of the active fuel region. Fast and thermal flux measurements were made for several of the cores.

A.10.2 Fuel Element Description

The SPERT III elements are plate type with many characteristics similar to the APPR elements (SM-1, SM-1A, PM-2A, and SM-2). The major difference between the SPERT III elements and the APPR type is the inclusion of a stainless steel reinforcing plate through the center of the element, perpendicular to the fuel plates. The SPERT III core contain stationary elements as well as two different types of control rod fuel elements. The physical and geometrical characteristics of the elements are given in Table 4.

A.11 ANL EXPERIMENTS ON G.E. TYPE ELEMENTS

One of the first experiments on the subject type cores was performed by ANL in conjunction with KAPL. (36)

A.11.1 Scope of Experiments

Seven critical masses containing no boron and with control rods completely withdrawn were obtained. Six of these configurations were associated with studies on the effects of various iron reflectors. The seventh core was a minimum critical mass assembly. Two other critical cores were constructed in which one control rod was fully inserted. Fast and thermal flux plots were obtained for several of the cores.

A.11.2 Fuel Element Description

The fuel elements were composed of 93.15% enriched uranium foil, encased in polyethylene sheets, and attached to steel plates. The active portion of the element was 20 in. long, with a 1 in. square cross section.

APPENDIX B - TRANSIENT EXPERIMENTS

The correlation of reactor parameters need not be restricted to static conditions. A number of experiments have been conducted to determine the characteristics of fully enriched UO_2 , stainless steel, light water moderated cores under rapid transient conditions. Most of this work has been performed by Phillips Petroleum at the SPERT I and SPERT III facilities. This includes experiments on the SM-1 ZPE Core I, the BSR-II, and SPERT III cores. These experiments are discussed below.

B.1 SM-1 ZPE CORE I ELEMENTS IN SPERT I

An extensive series of tests has been conducted on a core composed of 19 SM-1 ZPE Core I elements (see Section A.2.2). Included were power excursions with reactor periods varying from 0.005 to 600 msec. Data is available from tests initiated from room temperature (37), (38), (39), and from near-saturation temperatures (34) at atmospheric pressure. Peak power and total energy release to time of shutdown is tabulated as a function of inverse reactor period. Also of interest in transient studies are the ℓ/β_{eff} determinations made on the core (40). Reactivity oscillator experiments have also been performed on this core.

B.2 BSR-II

Rapid transient data, similar to that obtained on the SM-1, is available for the BSR-II. The most rapid transients investigated with this core had a reactor period of 0.015 msec. Measurements of ℓ/β_{eff} were obtained by ORNL (32) as well as at the SPERT I facility (40).

B.3 SPERT III (41), (42), (43)

In the transient tests on the SPERT III core, initial system pressure was varied from atmosphere to 2500 psig. Coolant flow was varied from 0 to 18 ft/sec.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX C - REDUCTION OF EXPERIMENTAL DATA

The core configurations discussed in Appendix A and listed in Table 4 were restricted to systems which were within \$0.50 of criticality. Where possible, the exact critical mass was determined with the aid of U-235 reactivity coefficients. The number densities of U-235 reported in Table 4 were based on these extrapolated critical masses and the actual core volume as constructed.

The critical bucklings given in Table 4 were obtained with the aid of experimental fast flux traverses. The flux distributions in the slab cores were fitted with a cosine curve, and a J_0 distribution was assumed for those systems approximating a cylinder. Extrapolation of the cosine and J_0 curves to their first zero values yielded the required dimensions for the buckling calculations.

TABLE 4

PHYSICAL GEOMETRICAL PROPERTIES OF CRITICAL CORES

Experiment By	Core	Core Description	Equivalent Core Radius, cm	Active Core Height, cm	Critical Mass of U-235, kg	B-10 Content, gm	Metal Fraction	Atomic Concentrations, Atoms/cc x 10 ⁻²⁰				Buckling, cm ⁻² x 10 ⁴
								U-235	B-10	SS	H	
ANL	SM-1	GE type fuel ele.	17.6	50.8	8.087	0	0.30	4.103		255.	491	124
ORNL	SM-1	7x7 core, corners removed	28.2	55.88	10.5	0	0.197	1.77	0	168	537	70.9
ORNL	SM-1	7x7 core, corners removed	28.2	55.88	9.9	0	0.182	1.83	0	155	547	70.9
ORNL	SM-1	7x7 core, corners removed	28.2	55.88	6.6	0	0.095	1.21	0	80.9	605	70.9
ORNL	SM-1	5x5 core	21.0	55.88	4.6	0	0.106	1.52	0	90.2	598	100.8
ORNL	SM-1	5x5 core	21.0	55.88	7.4	0	0.197	2.45	0	168	537	100.8
ORNL	SM-1	7x7 corners removed zero void vol.	28.2	55.88	10.35	0	0.197	1.90	0	168	537	70.9
ORNL	BSR-2	20 full ele. and four 1/4 el. and one 1/2 el.	19.3	38.1	5.95	0	0.227	3.10	0	193	517	124.2
Alco	SM-2	7x7 corners removed	28.2	54.9	6.02	0	0.06578	1.165	0	55.7	625	71.5
Martin	ERDL	No boron, no S.S. core	11.	55.88	3.81	0	0.01	3.93	0	0	662	201.9
Martin	ERDL	211 full plates	17.3	55.88	8.12	0	0.245	3.96	0	209	505	128.0
Martin	ERDL	211 full plates and 523 half plates	28.2	55.88	13.25	0	0.181	2.43	0	154	548	70.9
Martin	PM-1	Min. size crit. core	10.3	76.2	5.011	0	0.150	5.06	0	128	569	230.4
Martin	ANPP	3x3 core No. 450	12.6	55.88	8.864	0	0.277	8.15	0	236	484	187.0
Martin	ANPP	3x2 - 7/18 core No. 350	11.3	55.88	7.085	0	0.210	8.10	0	179	529	200.
Martin	ANPP	3x2 -14/18 core No. 400	12.1	55.88	8.234	0	0.254	8.21	0	216	499	192.
Martin	ANPP	3x2 9.5/18 core No. 350	11.6	55.88	7.470	0	0.210	8.10	0	179	529	208.
Phillips Pet.	SPERT I	3-row slab with SPERT Type 15 fuel ele.	15.3	91.4	8.290	0	0.23	3.16	0	196	515	136.
Phillips Pet.	SPERT II	Smallest critical core	22.0	91.4	14.210	0	0.23	2.62	0	196	515	83.0
Phillips Pet.	SPERT II	Smallest critical core with S.S. filler pcs. in reflector region	22.4	91.4	14.920	0	0.23	2.65	0	196	515	80.9
Martin	ANPP	3x3 core No. 200 3 plastic spacers per ele.	12.63	55.88	4.422	0	0.154	6.499	0	123.5	565.9	196
Martin	ANPP	4x4 core No. 440	16.35	55.88	18.204	0	0.399	9.377	0	321.99	402.0	154
Alco	SM-2	Min. size 3x3 -18 plates per ele. and 16 plates per c.r. ele.	12.63	54.9	7.189	0	0.218	6.83	0	186	486	171
Martin	ANPP	5x3 21.1/22 core No. 640	18.726	55.88	23.857	0	0.4695	9.933	0	392.7	355	110.3
Martin	ANPP	6x6-1.12/4 core No. 4302	25.836	55.88	12.45025	0	0.2034	2.723	0	168	532.8	80.
Martin	ANPP	5x5-2.6/12 core No. 4803	21.494	55.88	8.56959	0	0.2034	2.708	0	168	532.8	99.

TABLE 4 (CONTINUED)

Experiment By	Core	Core Description	Equivalent Core Radius, cm	Active Core Height, cm	Critical Mass of U-235, kg	B-10 Content, gm	Metal Fraction	Atomic Concentrations, Atoms/cc x 10 ⁻²⁰				Buckling, cm ⁻² x 10 ⁴
								U-235	B-10	SS	H	
Martin	ANPP	5 x 4 core No. 4804	18.822	55.88	6.61842	0	0.2034	2.727	0	168	532.8	117.
ANL	GE	Clean slab, 170 elements	25.4 x 19.55	50.8	9.044	0	0.30	3.454	0	255.6	474	110.6
ANL	GE	Slab with 1" steel reflector on one side	25.4 x 19.85	50.8	9.31	0	0.30	3.454	0	255.6	474	112.7
ANL	GE	Slab with 2" steel reflector on one side	25.4 x 19.55	50.8	9.044	0	0.30	3.454	0	255.6	474	114.2
ANL	GE	Slab with 3" steel reflector on one side	25.4 x 18.94	50.8	8.778	0	0.30	3.454	0	255.6	474	115.7
ANL	GE	Slab with 4" steel reflector on one side	25.4 x 18.74	50.8	8.672	0	0.30	3.454	0	255.6	474	116.4
ANL	GE	Slab with 6" steel reflector on one side	25.4 x 19.44	50.8	8.512	0	0.30	3.454	0	255.6	474	117.4
Martin	ANPP	4 x 3-8/18 core No. 402	15.578	55.88	13.483	12.71	0.24925	8.111	0	196.91	502.4	141.
Martin	ANPP	5 x 4-5/18 core No. 404	19.875	55.88	22.019	42.91	0.25317	8.145	0	200.2	499.8	109.
ORNL	SM-1	7 x 7 core, corners removed	28.2	55.88	11.32	4.15	0.197	2.08	0.0179	168	537	70.9
ORNL	SM-1	7 x 7 core, corners removed	28.2	55.88	11.76	6.39	0.197	2.16	0.0275	168	537	70.9
ORNL	SM-1	7 x 7 core, corners removed	28.2	55.88	13.30	15.56	0.197	2.44	0.0671	168	537	70.9
ORNL	SM-1	7 x 7 core, corners removed	28.2	55.88	13.60	15.56	0.197	2.50	0.0671	168	537	70.9
ORNL	SM-1	7 x 7 core, corners removed	28.2	55.88	15.18	23.33	0.197	2.79	0.101	168	537	70.9
ORNL	SM-1	7 x 7 core, corners removed	28.2	55.88	16.63	31.11	0.197	3.05	0.134	168	537	70.9
ORNL	SM-1	7 x 7 core, corners removed	28.2	55.88	16.67	31.11	0.197	3.06	0.134	168	537	70.9
ORNL	SM-1	7 x 7 core, corners removed	28.2	55.88	18.52	38.89	0.197	3.40	0.168	168	537	70.9
ORNL	SM-1	7 x 7 core, corners removed	28.2	55.88	20.10	46.67	0.197	3.69	0.201	168	537	70.9
Alco	SM-1 ZPE	Minimum Critical Core	17.3	55.88	7.8	7.5	0.193 ss	3.80	0.0860	164	540	128.
Alco	SM-1 ZPE	7 x 7 core, corners removed	28.2	55.88	22.48	37.80	0.185	4.13	0.163	157	545	70.9
Alco	SM-1 ZPE	6 x 6 core, corners removed	23.8	55.88	15.80	19.54	0.187	4.07	0.118	159	544	86.6
Alco	SM-1 ZPE	5 x 5 core, corners removed	19.2	55.88	10.65	4.93	0.235	4.02	0.0459	163	541	113.3
Alco	SM-1 ZPE	7 x 7 core, corners removed	28.2	55.88	22.48	19.52	0.361	4.13	0.0842	266	460	70.9
Alco	SM-1 ZPE	6 x 6 core, corners removed	23.8	55.88	15.80	13.71	0.246	4.07	0.0830	240	480	86.6
Alco	SM-1 ZPE	5 x 5 core, corners removed	19.2	55.88	10.14	8.79	0.231	4.02	0.0818	197	514	113.
Alco	SM-1A	Minimum Critical Core	17.3	55.88	7.75	5.4	0.193	3.78	0.0619	164	540	128.
Alco	PM-2A	3 rodged Min. Crit. Core	16.2	55.88	7.76	6.1	0.193	4.32	0.0797	164	540	139.
Alco	SM-2	7 x 7 Corners removed	28.2	54.9	9.2	12.28	0.088	1.72	0.0539	75.0	610	71.5
Alco	SM-2	7 x 7 Corners removed	28.2	54.9	23.0	62.3	0.181	4.30	0.274	154	548	71.5
Alco	SM-2	7 x 7 Corners removed	28.2	54.9	22.1	58.2	0.169	4.13	0.256	144	556	71.5

TABLE 4 (CONTINUED)

Experiment By	Core	Core Description	Equivalent Core Radius, cm	Active Core Height, cm	Critical Mass of U-235, kg	B-10 Content, gm	Metal Fraction	Atomic Concentrations, Atoms/cc x 10 ⁻²⁰				Buckling, cm ⁻² x 10 ⁴
								U-235	B-10	SS	H	
Alco	SM-2	7 x 7 Corners removed	28.2	54.9	7.0	3.52	0.078	1.31	0.0155	66.4	617	71.5
Martin	ERDL	348 fuel plates	19.2	55.88	9.74	10.25	0.184	3.86	0.0954	157	546	113.
Martin	ERDL	357 fuel plates to 224 half	21.8	55.88	13.13	13.70	0.231	4.03	0.0989	197	514	96.3
Martin	ERDL	Third plastic spacer added to above core	18.0	55.88	12.77	13.31	0.227	3.92	0.0961	193	517	96.3
Martin	MPR	Minimum critical mass core	18.0	58.4	7.988	10.97	0.124 ss	3.17	0.1029	106	586	120.
Martin	PM-1	Core with 19, 0.27 wt. % of boron tube	12.6	76.2	6.876	7.28	0.183	4.64	0.115	156	547	180.
Martin	PM-1	Core with 17, 0.27 wt. % boron tubes	12.5	76.2	6.797	6.51	0.180	4.66	0.105	153	549	182.
Martin	PM-1	Core with 14, 0.27 wt. % boron tubes	12.2	76.2	6.521	5.36	0.177	4.69	0.0906	151	551	187.
Martin	PM-1	Core with 13, 0.27 wt. % boron tubes	11.8	76.2	6.163	4.98	0.176	4.74	0.0900	150	551	195.
Martin	ANPP	5 x 5 core No. 454	21.0	55.88	24.508	47.70	0.176	8.11	0.3437	216	489	99.
Martin	ANPP	4 x 4-8/18 core No. 453	17.7	55.88	17.507	24.64	0.270	8.16	0.2479	209	493	124.
Martin	ANPP	4 x 3-12/18 core No. 452	16.1	55.88	14.388	13.54	0.273	8.10	0.1657	214	489	139.
Martin	ANPP	5 x 4-1/18 core No. 354	19.2	55.88	19.832	38.59	0.219	7.85	0.3437	168	525	114.
Martin	ANPP	4 x 3-12/18 core No. 353	16.1	55.88	14.307	20.09	0.209	8.06	0.2479	161	531	137.
Martin	ANPP	4 x 3-3/18 core No. 352	15.0	55.88	12.392	11.69	0.214	8.04	0.1657	166	527	153.
Martin	ANPP	4 x 4 core No. 403	16.8	55.88	15.834	22.32	0.244	8.19	0.2479	190	508	127.
Phillips Pet.	SPERT	SM-1 ZPE core 18/19 cut mass	17.8	55.88	7.6	5.35	0.193 ss	3.50	0.0579	164	540	124.

TABLE 5

**SUMMARY OF SCOPE OF EXPERIMENTS PERFORMED ON LIGHT WATER MODERATED,
FULLY ENRICHED URANIUM, STAINLESS STEEL CORES**

Experiments	Critical Masses			Material Worths	Flux & Power Distribution	Reflector Effects
	Clean, no rods	borated no rods	rods fully inserted			
ANL on GE type elements	X		X		X	X
ORNL on SM-1 Mockup	X	X	X	X	X	
Alco on SM-1		X		X	X	X
Martin on ERDL	X	X		X	X	X
Martin on MPR		X		X	X	X
ORNL on BSR-II	X		X			
Phillips Petroleum on BSR-II					X	
Alco on SM-2	X	X	X	X	X	X
Alco on SM-1A		X			X	
Alco on PM-2A		X			X	
Martin on ANPP	X	X		X	X	
Martin on PM-1	X	X		X	X	
Phillips Petroleum on SPERT III	X		X		X	

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

REFERENCES

1. Owen, B. G., "An Empirical Correlation of The Experimental Data on Homogeneous Critical Assemblies of Uranium and Hydrogen of All Enrichments", PG Report 97(R), November 1960.
2. Paxton, H. C., "Critical Data for Nuclear Safety Guidance", LAMS-2415, May 16, 1960.
3. Paxton, H. C., "Correlations of Experimental and Theoretical Critical Data", LAMS-2537, May 15, 1961.
4. Brown, W. S., and Dixon, M. H., Project Engineers, "Fiscal Year 1962 Program Plan for Engineering Support and Development of Army Pressurized Water Reactor Power Plants", AP Note-378, September 6, 1961.
5. Bobe, P. E., "Interim Report of Nuclear Analysis Performed on SM-2 Core and Vessel - September 1, 1958 to December 31, 1959", APAE No. 65, May 27, 1960.
6. Williams, D. V. P., et. al., "Army Package Power Reactor Critical Experiment", ORNL-2128, August 21, 1956.
7. Noaks, J. W., editor, "Extended Zero Power Experiments on the Army Package Power Reactor ZPE-2", APAE No. 21, November 15, 1957.
8. O'Brien, J. F., Project Engineer, "ERDL Task 100, Phase 3, Zero Power Check Point Experiments", MND-E-1847, April 1959.
9. Boch, A. L., Project Coordinator, "Army Package Power Reactor Project Progress Report", ORNL 2703, April 6, 1959.
10. Pinkel, B., and Young, G. B. W., "A Discussion of the Correlation of Critical Conditions For Bare Homogeneous Reactors", P-1713, June 4, 1959.
11. Hellens, R. L., "Multigroup Fourier Transform Calculation, Description of MUFT-III Code", WAPD-TM-4, July 1956.

12. Cannon, G. A., et. al., "APWRC-CELCOR, A Fortram II Program for Multiple Lethargy Level P_1 or S_2 to S_{16} DSN Cell Corrections in Cylindrical Spherical, and One or Two Dimensional Slab Geometry", MND-C-2549, December 27, 1960.
13. Byrne, B. J., and Caton, R. L., "Two Dimensional P_3 Calculation for APPR-Type Fixed Fuel Elements", AP Note 96, February 14, 1958.
14. McElligott, P. E., "Fink-1: A Two Group, Two Region Calculation on the IBM-650", APAE Memo-121, January 28, 1958.
15. Noaks, J. W., and Johnson, W. R., "Army Package Power Reactor, Zero Power Experiments, ZPE-1, APAE No. 8, February 8, 1957.
16. Beaver, R. J., et. al., "Specifications for Army Package Power Reactor (APPR-1) Fuel and Control Rod Components", ORNL-2225, August 7, 1957.
17. 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 No. 58 Revised, October 12, 1960.
18. Raby, T. M., et. al., "PM-2A Core I Zero Power Equipment", APAE No. 75, October 21, 1960.
19. Noaks, J. W., et. al., "SM-2 Critical Experiments, CE-1" APAE No. 54, November 30, 1959.
20. McCool, W. J., et. al., "Extended SM-2 Critical Experiments CE-2", APAE No. 54, Supplement 1, June 30, 1961.
21. O'Brien, J. F., Project Engineer, "PM-1 Critical Experiments and Zero Power Testing", MND-M-1858, September 1961.
22. Akin, R. J., Project Engineer, "Nuclear Studies on the MPR Zero Power Test Core", MND-MPR-1646, December 23, 1958.
23. Olsen, T. M., et. al., "ANPP Core Development Program Pressurized Water Task Quarterly Progress Report Number One", MND-C-2200, November 1959.
24. Olsen, T. M., et. al., "ANPP Code Development Program Pressurized Water Task Quarterly Progress Report Number Two", MND-C-2201, February 1960.

25. Olsen, T. M., et. al., "ANPP Core Development Program Pressurized Water Task Quarterly Progress Report Number Three", MND-C-2202, May 1960.
26. Olsen, T. M., et. al., "ANPP Core Development Program Pressurized Water Task Quarterly Progress Report Number Four", MND-C-2203, August 1960.
27. Olsen, T. M., et. al., "ANPP Core Development Program Pressurized Water Task Quarterly Progress Report Number Five", MND-C-2204, November 1960.
28. O'Brien, J. F., Project Engineer, "ANPP Core Development Program Pressurized Water Task, Quarterly Progress Report Number Six", MND-C-2205, February 1961.
29. Eicheldinger, C., Project Manager, "ANPP Core Development Program, Pressurized Water Task, Quarterly Progress Report Number Seven", MND-C-2206, May 1961.
30. Eicheldinger, C., Project Manager, "ANPP Core Development Program, Pressurized Water Task, Quarterly Progress Report Number Eight", MND-C-2207, August 1961.
31. Eicheldinger, C., Project Manager, "ANPP Core Development Program, Pressurized Water Task Progress Report Number Nine", MND-C-2208, February 1962.
32. Blizzard, E. P., Director, "Neutron Physics Division Annual Progress Report for Period Ending September 1, 1959", ORNL-2842, November 16, 1959.
33. Schroeder, F., editor, "Quarterly Technical Report, SPERT Project, October, November, December, 1959", IDO-16616, August 24, 1960.
34. Haire, J. C., editor, "Quarterly Technical Report, SPERT Project, April, May, June, 1959", IDO-16584, April 12, 1960.
35. Schroeder, F., et. al., "The SPERT III Reactor Nuclear Startup", IDO-16586, March 18, 1960.
36. Martens, F. H., and Helfrick, G. F., "Critical Studies With G. E. Type Fuel Elements", ANL-5108, August 1953.

37. Haire, J. C., editor, "Quarterly Progress Report, SPERT Project, April, May, June, 1959", IDO-16584, April 12, 1960.
38. Bentzen, F. L., editor, "Quarterly Progress Report, SPERT Project, October, November, December, 1958", IDO-16537, September 1, 1959.
39. Norberg, J. A., editor, "Quarterly Progress Report, SPERT Project, January, February, March, 1959", IDO-16539, November 20, 1960.
40. Spano, A. H., "Perturbation Measurements of α/β_{eff} Using A 1/ Absorber", IDO-16638, March 3, 1961.
41. Schroeder, F., editor, "Quarterly Technical Report, SPERT Project, April, May, June, 1961", IDO-16716, September 15, 1961.
42. Schroeder, F., editor, "Quarterly Technical Report, SPERT Project, October, November, December", IDO-16687, June 1, 1961.
43. Schroeder, F., editor, "Quarterly Technical Report, SPERT Project, July, August, September, 1961", IDO-16726, December 15, 1961.