DEPARTMENT OF ENERGY
NAVAL REACTORS LABORATORY FIELD OFFICE
POST OFFICE BOX 109
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NRLFO:OC:22-020
18 April 2022

Dear Mr. Aftergood:

This constitutes a final response to the request for information that you submitted electronically to the United States Department of Energy (DOE) under the Freedom of Information Act (FOIA), 5 U.S.C. §552. Your request was filed with the DOE Headquarters FOIA Office and assigned tracking number HQ-2018-01232-F. It was then transferred to this office for response and assigned tracking number NRLFO-2018-01343-F.

You requested, on behalf of the Federation of American Scientists, "[A] copy of this 1959 technical report that was produced by the Atomic Energy Commission: Naval Reactors Physics Handbook, Volume 2. The Physics of Pressurized Water Reactors. Edited by Sidney Krasik, February 1959."

You further noted that "[a]t the time it was published in 1959, this document was classified Confidential. (Volumes 1 and 3 of the Handbook are unclassified and have previously been released.) If it is still classified today, we request that it be reviewed for declassification."

You received interim response NRLFO_OC_18-017 dated 13 July 2018, which noted that the document you requested had been located and was at that time classified Confidential. The interim response further noted that the Naval Reactors Program would conduct a declassification review and/or segregability analysis on this document, and that a final response to your request could not be made until such work was complete.

The Naval Reactors Program has completed its declassification review and segregability analysis of the requested document, and an unclassified form of the document, with classified information redacted, is being released to you in its entirety. The redacted portions of this document are withheld from release under the authority of the Atomic Energy Act, 42 U.S.C. §§2162, 2167, 2168(a)(1), as they comprise Restricted Data within the meaning of that Act. The Atomic Energy Act is a qualifying statute for purposes of Exemption 3 of the FOIA.

You have been categorized as representing a non-commercial scientific institution under the U.S. Department of Energy regulation implementing the FOIA, based upon the definition at 10 CFR 1004.2(k). Any fees applicable to the foregoing effort would ordinarily be addressed in accordance with 10 CFR 1004.9. However, the cost of responding to your request was less than $15.00, the amount DOE incurs to process a fee collection; in accordance with 10 CFR 1004.9(a)(6), no fees are due.

Mr. Matthew J. Brott is the authorizing official for purposes of determining responses to FOIA requests. This response may be appealed within 90 calendar days from your receipt of this letter pursuant to 10 C.F.R. § 1004.8. Appeals should be addressed to Director, Office of Hearings and Appeals, HG-1, L'Enfant Plaza, U.S. Department of Energy, 1000 Independence Avenue, S.W., Washington, D.C. 20585-1615. The written appeal, including the envelope, must clearly indicate that a FOIA appeal is being made. You may also submit your appeal to OHA.filings@hq.doe.gov, including the phrase "Freedom of Information Appeal" in the subject line. The appeal must contain all of the elements required by 10 C.F.R. § 1004.8, including a copy of the determination letter. Thereafter, judicial review will be available to you in the Federal District Court either: 1) in the district where you reside; 2) where you have your principal place of business; 3) where DOE's records are situated; or 4) in the District of Columbia.

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I appreciate the opportunity to assist you with this matter. If you have any questions about this correspondence, please contact me at (412) 476-7202.

Sincerely,

C. P. Nunn
FOIA Coordinator

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1ST REVIEW – DATE:	3/8/2022	DETERMINATION [CIRCLE NUMBER(S)]	
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	Stephen T. Bell, Chief Physicist, NR	(3) CONTAINS NO NNPI	
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NAME:	Laura J Nanko	(7) OTHER (SPECIFY):	Classified Information Redacted
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~~CONFIDENTIAL~~

THE PHYSICS OF PRESSURIZED WATER REACTORS

VOLUME 2, BOOK 1, NAVAL REACTORS PHYSICS HANDBOOK

VOLUME EDITOR
S. KRASIK

Manager, Central Physics and Mathematics
Westinghouse Electric Corporation
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NAVAL REACTORS BRANCH, DIVISION OF REACTOR DEVELOPMENT
UNITED STATES ATOMIC ENERGY COMMISSION

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This handbook is one of a series sponsored by the Naval Reactors Branch of the Atomic Energy Commission to publish in useful form the technology being developed in the Naval and Shippingport (PWR) Reactor Programs. This series includes:

Liquid Metals Handbook
First edition: Edited by R. N. Lyon, June 1950
Second edition: Edited by R. N. Lyon, June 1952
Third edition: (Sodium-NaK Supplement) Edited by C. B. Jackson
First printing June 1955; second printing (Revised), November 1955 (Available from Superintendent of Documents, Washington 25, D.C.)

Metallurgy of Zirconium. Edited by B. Lustman and F. Kerze, Jr., July, 1955 (Published by McGraw-Hill Book Company, New York)

The Metal Beryllium. Edited by D. W. White and J. E. Burke, July 1955 (Published by American Society for Metals, Cleveland)

Bibliography of Reactor Computer Codes, Report AECU-3078. Edited by R.S. Brodsky, December 1955 (Available from Superintendent of Documents, Washington 25, D.C. This compilation is being kept current by the Nuclear Codes Group quarterly newsletter. Inquiries should be sent c/o: A.E.C. Computing Facility, New York University, New York, N.Y.)

Reactor Shielding Design Manual. Edited by T. Rockwell, III, March 1956 Published separately by Government Printing Office, McGraw-Hill Book Company, New York, and D. Van Nostrand Company, Princetown, N.J.)

Corrosion and Wear Handbook for Water-cooled Reactors. Edited by D.J. DePaul, March 1957 (Published separately by Government Printing Office and McGraw-Hill Book Company, New York)

Naval Reactors Physics Handbook. A. Radkowsky, Chairman of Editorial Board.
Volume I The Physics of Naval Reactors; Basic Techniques. Edited by A. Radkowsky, in preparation
Volume II The Physics of Pressurized Water Reactors. Edited by S. Krasik
Volume III The Physics of Intermediate Spectrum Reactors. Edited by J.R. Stehn, September 1958

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Reactor Plant Piping Handbook. Edited by M. Shaw, in preparation

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Nuclear Poison Materials for Reactor Control. Edited by W.K. Anderson and J.S. Theilacker, in preparation

FOREWORD

The Naval Reactors Physics Handbook, which is the seventh in a series of handbooks on the basic reactor technology developed under the sponsorship of the Naval Reactors Branch of the Atomic Energy Commission, presents the analytical and experimental physics techniques which have been developed for the reactors designed in the Naval and Shippingport (PWR) Reactor Programs. Reactor physics has been strongly emphasized in these programs, and extensive experimental and computational facilities have been developed. Further, particular attention has been paid to obtaining from reactor operation as much physics information as possible. Prototype reactors have been carefully instrumented and elaborate series of physics tests have been carried out with the two-fold objectives of confirming and refining the physics calculations used in each particular reactor design and of obtaining increased general understanding of reactors.

Power reactors must meet many stringent requirements such as size limitations, stability, and reliability. Above all, changes cannot be made in a finished core to compensate for design error. The fundamental importance of physics in establishing fuel loading, start-up, control, power distribution, and endurance is widely recognized. What is perhaps less well-realized is the constant interplay of mechanical, metallurgical, and plant considerations with physics to provide a well-balanced design. The requirements on physics are therefore especially severe in a power reactor. In order to provide precise and dependable physics information it has been necessary to develop many detailed methods of analysis. Many of these analytical techniques are applicable to other reactors of widely differing characteristics.

The physics methods reported in these volumes are characterized by their having been found useful in the design of actual power reactors. In addition, basic reactor theory developed in the laboratories participating in the Naval Reactors Program is also covered. This includes many areas of reactor physics which have long presented particular difficulty, for example thermal neutron spectra, control rod effectiveness, self-shielding and transport effects, temperature coefficients of reactivity, and flux stability.

While I am pleased with the advances in physics which have been made in the course of the Naval Reactors and Shippingport Programs, I know that there is a great deal more that must be done to improve our understanding of reactors and our ability to design them for longer life, more uniform power distribution, simpler control, and more effective utilization of the fuel. I hope that by publishing our available data and techniques in these volumes we will provide impetus toward the solution of many of the basic reactor physics problems still outstanding.

H. G. RICKOVER
Chief, Naval Reactors Branch
Reactor Development Division
U. S. Atomic Energy Commission

EDITORIAL BOARD PREFACE

The purpose of this handbook is to present the most pertinent parts of the body of physics knowledge which has been built up in the course of the Naval and Shippingport (PWR) Reactor Programs, with the aim of providing a background of understanding for those interested in nuclear core design. Much of the material has already been published either in scientific journals or topical laboratory reports; scientists engaged in the Naval Reactors Program, as elsewhere, are encouraged to report their work in the normal manner. However, as is usual in rapidly expanding programs, much data have remained in the form of internal memoranda and other informal communications. It was felt to be desirable and timely to review and recapitulate the work which has been done on the Program and to make it available in convenient and readily applicable form. Wherever practicable the chapters in these volumes have been prepared by authors who actively participated in the developments discussed.

Volume I of the handbook attempts to bring together the basic theoretical and experimental material which is of especially wide interest or is common to both thermal and intermediate-neutron-energy reactor types. The physics design of light-water-moderated and -cooled reactors is covered in Volume II, and that of intermediate-neutron-energy power reactors in Volume III.

One of the rewards of editing a work of this kind has been the discovery of many illuminating relationships between the development programs of different types of reactors. For example, in an intermediate reactor in which the physics characteristics in the core do not change greatly with temperature it was natural in the critical assembly work to rely heavily upon a quite accurate mock-up of the core at room temperature. In light-water reactors, the philosophy was somewhat different. Since the physics properties of the core change greatly with temperature, an accurate mock-up at room temperature was not so important; the ability of calculational procedures to predict the results of the critical experiments at room temperature was reasonable assurance that the calculational procedures would also be valid under power operating conditions. This has indeed proved to be the case. Even so, as performance requirements have become more severe, it has been necessary to go to more and more faithful mock-ups of water-cooled reactors. Along with these developments, analytical techniques, aided by large digital computers, have been making great advances.

Reactor physics today is such a rapidly growing field that the book is necessarily uneven since it has been prepared over the course of several years by editors operating on a part-time basis. All members of the Editorial Board have participated fully in this work by reading all chapters and providing detailed comments. Criticism and suggestions by the reader will be welcomed.

The writer wishes to express his appreciation to Rear Admiral H.G. Rickover for suggesting the preparation of this handbook and for giving much encouragement at all times. Special thanks are also due to T. Rockwell, III, Technical Directory of the Naval Reactors Branch, for many helpful suggestions.

A. RADKOWSKY
Chairman, Editorial Board
Naval Reactors Physics Handbook

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Chapter 1

INTRODUCTION

S. Krasik

1.1 PURPOSE AND CONTENT OF VOLUME II

The purpose of this volume of the Naval Reactor Physics Handbook is to present the methods currently used in the nuclear design of pressurized water reactors. This type of reactor presently occupies a dominant position in the field of marine propulsion. In central station power generation it will, in all probability, occupy a significant position for a long time. Because of its importance, there is a wide interest in the techniques by which the nuclear design of the pressurized water reactor is established. These techniques have resulted from the contribution of many individuals and many laboratories over the past ten to fifteen years. While essentially all of the material is available in various topical reports, this is the first attempt to collect and present it conveniently in one volume, emphasizing its application to design procedure.

There is not available at this time a standardized analytic design procedure for pressurized water reactors. That is to say, there is not a calculational procedure through which the properties of the operating power reactor can be calculated with sufficient certainty for design needs. This deficiency has made it necessary to establish certain crucial features of the design by experiment.

The methods of nuclear design for pressurized water reactors are in a state of rapid development. In this volume, methods and techniques in current use only are described. The method of presentation is primarily through the analysis of certain reactor designs, selected from those developed under the cognizance of the Naval Reactors Branch of the Atomic Energy Commission. The applied procedures given in this volume are not the ones which were actually used to design certain of these reactors. This is most obvious in the case of the S1W (formerly designated the STR Mark I). It would serve little purpose to present here design methods which have been displaced by newer techniques. For this reason, the analyses of all reactor designs have been considered only from the point of view of current procedures, and the evaluation of experimental and operational data have been subjected to the same limitations.

This volume of the Handbook is organized into six parts. Part I deals with the nuclear design procedure for small, highly-enriched reactors. The S1W reactor cores are used as the

expository vehicle. In effect, this section represents an analysis of the SIW reactor cores in terms of current design procedures.

Part II is a design status report on other selected reactor projects which are under the supervision of the Naval Reactors Branch. The PWR is presented as an example of a large, high-uranium-content reactor designed primarily for central station power production. The next reactors discussed are the AIW designs. [redacted] The other two reactors described in this section are submarine reactors. [redacted]

Part III of this volume deals with pressurized water reactor experimentation. The general design requirements for critical experiments are discussed, as well as the specific designs of several critical experiments. The techniques used in critical experimentation are described and typical results are presented. The subject of flux mapping is treated in detail; the measurement of resonance escape probability, fast fission factor and thermal utilization are also discussed.

In Part IV, certain reactor experimental results of particular significance to basic reactor theory are presented and compared with analysis. Clean, critical data for both highly-enriched reactors and slightly-enriched reactors are included. In addition, the available data from control rod worth experiments are given. The general procedure in the design of pressurized water reactors is to compare experimental results on cold, critical experiments with a theoretical analysis of the cold experiments. The theory is then extrapolated to the hot case. The particular importance of basic experiments in pressurized water reactor design is a consequence of this procedure.

Part V is a section on the topic of reactor and plant kinetics. Because of the negative temperature coefficient of reactivity, pressurized water reactors are coupled in response to the entire plant. The basic analysis of the coupled problem is presented, together with application to the problem of reactor and plant stability.

In Part VI, two special topics of importance to the design of pressurized water reactors are discussed. One of these is the theory of control rod worth, insofar as this topic has been developed to date. The other topic deals with synthesis methods for calculating reactor designs.

1.2 THE PROBLEMS OF NUCLEAR DESIGN

There are three major questions in the nuclear design of a power reactor which must be answered:

1. Is the control system adequate to assure shutdown of the reactor in its most reactive condition?
2. Does the reactor have sufficient excess reactivity to meet its lifetime and maneuvering requirements?
3. What is the power distribution in the reactor throughout its lifetime?

It is clear that the first two questions are related, for the greater the excess reactivity of the reactor the greater is the total control requirement necessary to assure safe shutdown. It is perhaps not quite so clear that the third question is also related to the first two. This

comes about because of the important effect of the control rods on the power distribution in the reactor.

These questions are discussed in greater detail below. Before proceeding to this discussion, a few points will be noted relating to the general nuclear design problem. It is a characteristic of pressurized water reactors that the nuclear design is not only difficult conceptually, but complicated as well. There are two reasons for this. The first is that the reactor behavior is sensitive to relatively minor mechanical design details. The presence of water channels for control rods, the concentration of fuel into clusters, the water temperature, these all have significant effects on the reactor behavior. This sensitivity is related to the effective moderation properties of water and to the relatively short diffusion length for thermal neutrons in the reactor core. Hence, the neutron flux in the reactor can have extremely large gradients which affect both the criticality and power distribution in the reactor.

The second reason for the difficulty in the design of pressurized water reactors is related to experimentation with the reactor at its operating temperature. Experimentation under these conditions has presented such formidable problems that it has not yet been attempted. The only data available on the reactors at operating temperature have come from actual operation of the reactors. This difficulty has precluded an empirical design approach for pressurized water reactors, and has necessitated a substantial dependence upon theoretical methods for nuclear design. This approach includes extrapolation of experimental results from the cold condition to the reactor at operating temperature.

1.3 REACTIVITY AND CONTROL DESIGN

[redacted] The minimum acceptable shutdown for the reactor in its most reactive condition is approximately 5 per cent. [redacted]

To meet these needs the analytical methods available to the nuclear designer may be relied upon to the following extent:

1. The criticality of a uniform, homogeneously-fueled cold reactor may be calculated to within 2 per cent in reactivity.
2. [redacted] If the effects present in a given design are all recognized and taken into account, they may be calculated with a precision of 1 per cent to 2 per cent in reactivity.
3. [redacted]

4. [redacted] The accuracy of this calculation depends upon the reactor lifetime and decreases sharply with increasing reactor lifetime. The error can be one-quarter to one-half the total lifetime reactivity allotment, or 2 per cent to 5 per cent in reactivity.

Clearly, the accumulated uncertainty in an analytic reactor design can be very substantial. The total uncertainty is, in fact, so great as to preclude a completely analytic design if firm performance requirements are imposed on the reactor. For this reason, reactor critical experiments must be performed to establish the reactor design, even though these experiments impose a substantial burden in cost and time. Every reactor design developed in the Naval Reactor program has had to make use of a critical experiment to obtain the required degree of assurance in reactor performance.

1.4 ROLE OF THE MOCK-UP CRITICAL EXPERIMENT IN REACTIVITY AND CONTROL DESIGN

The two principal questions with respect to the reactivity and control design of the reactor are (1) adequate shutdown and (2) adequate excess reactivity. The adequacy of the reactor shutdown can be determined directly by constructing a critical experiment, which is a nuclear mock-up of the reactor design, and measuring the shutdown. Since the mock-up is a nuclear model of the design, this procedure is quite satisfactory.

The adequacy of the excess reactivity is not easily determined from experiment. As noted previously, the reactor cannot be easily mocked up in its high temperature condition. Furthermore, even if the hot reactor could be mocked up, the questions of xenon override and lifetime reactivity changes would still remain. Because of the primary difficulty of achieving a hot mock-up experiment, there has as yet been no consideration in the Naval Reactor program of mock-up of xenon or depletion conditions for pressurized water reactors.

Since mock-up critical experiments in the operating condition have not as yet been available, the approach employed, as noted previously, has been that of establishing the adequacy of the excess reactivity by calculation. This imposes a severe burden on the theoretical procedure, and requires that some assessment be made of the adequacy of the analysis. The procedure followed is to carry out analyses of the cold mock-up experiment in a variety of conditions. To the extent that the design analytic procedure appears adequate in explaining the behavior of the cold reactor, it may be relied upon for estimating the adequacy of the excess reactivity margin.

A variety of tests are imposed on the analysis of the cold mock-up experiment. These include simple "clean critical" conditions, a variety of geometries, specific heterogeneity effects, poison experiments, partial water height experiments, and organic moderator experiments. Considerable ingenuity has been shown in developing tests of the analytic procedures, and some significant weaknesses have been uncovered. While this design approach appears cumbersome, it has added in an important way to the understanding of reactor nuclear design.

1.5 SUMMARY OF REACTIVITY AND CONTROL DESIGN PROCEDURE

From the nuclear designer's point of view, the reactivity and control design procedure for a reactor project can be summarized in the stages listed below:

1. The objectives of the reactor project are set. Following this, initial survey studies are carried out to examine the possibilities of meeting the project objectives.
2. A general design area is selected. Based upon the results of the survey study, a more limited area is defined, and this is explored analytically in greater detail.
3. The general features of the reactor design are selected. Determination of the mechanical design permits work to proceed on the mock-up critical experiment. Detailed design studies are carried on during the period of procurement and construction of the mock-up critical experiment.
4. The essential mock-up critical experiment results are obtained and compared with calculations. If the design is reasonably satisfactory, a fuel loading is selected and core fabrication is undertaken.
5. The reactor core is manufactured and the plant placed in operation. During the manufacturing stage, confirmatory studies of the design are carried out. Following reactor operation, performance-analysis studies are made.

The steps outlined above are essentially those followed for each project under Naval Reactor cognizance, with the addition that, wherever possible, results of one project are applied to a new project. This procedure places greatest emphasis on certainty of the design. It is illustrated by the introduction of the critical experiment at as early a stage in the project as is feasible, and by the continuation of the critical experiment work through the manufacturing stage of the reactor project.

In addition to assurance of performance, another important objective is reduction of the over-all time required from concept of the reactor to initial plant operation. This objective is met by carrying out the detailed reactor design during construction of the mock-up critical experiment, and by proceeding to core fabrication with only a minimum confirmation of the design by the critical experiment. In every step, time is saved by proceeding to the next stage on the presumption that the design is satisfactory. The required confirmatory studies are carried out in parallel with the next stage of the project.

There is, of course, no unique design procedure for the nuclear design of a reactor. The procedure which is selected must balance the relative factors of available design time, certainty of performance, and over-all cost.

In concluding this summary of the reactivity and control design procedure, it is of interest to examine the freedom actually available to the nuclear designer in fixing a reactor design under this procedure. The nuclear designer has, first, the problem of obtaining an adequate range of control and, second, the problem of fixing the criticality of his design under given conditions at an appropriate point on this control range. The range of control available to the designer is severely limited by mechanical design considerations and the neutron absorption properties of materials suitable for use as control elements. Once the mechanical design of the reactor is fixed, and the number of control rods together with their span chosen, the nuclear designer's problem is reduced to determination of the fixed range of control available to him, rather than a choice of appropriate range of control. [redacted] However, once this choice is made, the nuclear designer can no longer exert significant influence on the reactivity and control design. His task then becomes the determination of design properties as they exist. Insofar as the nuclear designer is

concerned, the fixing of the reactor fuel loading and the start of core fabrication represent a major landmark in the reactor project's history.

1.6 POWER DISTRIBUTIONS IN THE REACTOR

Up to this point, the discussion of the nuclear designer's task has been concerned primarily with the reactivity and control design of the reactor. There remains the important problem of power distribution in the reactor. As noted previously, the power distribution is closely related to the control scheme since the control rods have a very significant effect on the power distribution. This represents another limitation on the amount of excess reactivity which the reactor may have, since a very large excess reactivity, even if controllable, may lead to severe power maldistributions in the reactor.

Control plays another important role in reactor power distributions. The rate of fuel depletion in the neighborhood of control rods is substantially below the average in the reactor. With high average depletion, this can result in a significant difference in fuel concentration between regions adjacent to control rods and the remainder of the reactor. As the reactor's lifetime nears an end, and the control rods are withdrawn, the flux depression due to the control rods is replaced by a flux peak due to the residual water channels previously occupied by the rods. This flux peak alone can be serious. In combination with the now relatively higher fuel density near the control rods it can result in severe local maldistributions of power in the reactor. [redacted]

The basic analytic tool available to the designer for these studies is the few-group diffusion theory. This is applied to design studies through the use of large scale digital computers.

The adequacy of diffusion theory as an approximation to transport theory for the description of the power distribution in the reactor has been in question for a number of years. Here again, the critical experiment provides important practical assistance. Flux distributions are made on the mock-up critical experiments and compared with analytic results. Once more, the point of view adopted is that a satisfactory analysis of the distribution in the cold mock-up reactor is reasonable assurance of adequacy for the operating reactor. It should be noted that the errors grow with lifetime, since errors in flux distribution calculations lead to errors in estimates of fuel depletion.

1.7 THE EVOLUTION OF REACTOR PHYSICS AND NUCLEAR DESIGN METHODS

It was remarked in the introduction to this chapter that nuclear design techniques for pressurized water reactors are in a state of rapid evolution. It was, in fact, noted that the design methods discussed in this volume were not available for the design of some of the reactors which are discussed. This evolutionary process is continuing because of the very real need of the nuclear designer for more adequate methods.

There are several areas in which advances will undoubtedly be made. One of these is development of more satisfactory methods of estimating control rod worth. This development can lead to a greatly reduced importance of the mock-up critical experiment in the design procedure. Another area in which significant advances will probably be made is in the calculation of the reactivity lifetime problem.

The advances in nuclear design techniques have come about in two principal ways. One is a better understanding of reactor physics. This has resulted from the gradual accumulation of analyzable experimental reactor information with which the theoretical results may be compared. The second way in which nuclear design techniques have advanced is through the application of large scale digital computers to reactor problems. This has made it possible to take into account the details of reactor composition and geometry which are of such importance in water moderated reactors. The next few years will see the continuation of this trend with the application of even larger digital computers as they become available.

1.8 ACKNOWLEDGEMENTS

In a field to which so many individuals and laboratories have contributed, the question of credit for original contributions is a difficult one. A few remarks should be made, however, with respect to authorship of the sections of this volume. The individual authors who have written the various sections of this book have been selected on the basis of their background in the topic and their availability for work on this handbook. Authorship of a section does not, of itself, represent a claim on the part of the author for original responsibility. Insofar as possible, credit for original work is given by reference to papers and reports.

The role of the national laboratories in the development of the Naval Reactor Program should also be mentioned. A major contribution in the area of presssurized water reactors for Naval Reactor propulsion was made by the Argonne National Laboratory in the program which led to the S1W reactor. The work which led to the S1W project had its inception at the Oak Ridge National Laboratory in 1946-1947 as an outgrowth of work which had been undertaken on the so-called Daniels power pile. The work was carried sufficiently far at Oak Ridge to permit a conceptual design report to be written.[1] [redacted] During 1948-1949 the Naval Reactor program and a large part of the personnel associated with it at Oak Ridge were moved to Argonne National Laboratory.

The next definitive landmark was the design status report[2] issued in 1950. The mock-up critical experiment ZPR-1 was placed in operation late in 1950, and the reactor fuel loading for the first core was set in April 1951. The critical experiment work at Argonne continued until 1953, and is reported in three documents.[3] The Bettis organization, which was formed in 1949, participated in the program with increasing responsibility and, following the setting of the first core fuel loading in 1951, the Argonne responsibility in the nuclear design was gradually shifted to Bettis. The design effort on a high performance, small, pressurized water reactor was started at the Knolls Atomic Power Laboratory in 1952.

Another laboratory which has been of considerable assistance to the pressurized water reactor development program is the Brookhaven National Laboratory. The cooperative study program on slightly enriched uranium water lattices represented an important contribution, particularly to the PWR project.

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1. H. Etherington, Study of a Water-Cooled Pile for Naval Application, Report ORNL-133, September 1948.
2. H. Etherington et al, Interim Report on Water-Cooled, Water-Moderated Reactors for Naval Application, Report ANL-4393, March 1950.
3. R. O. Brittan et al, Experiments on Zero Power Reactor, Report ANL-4684, August 1951.
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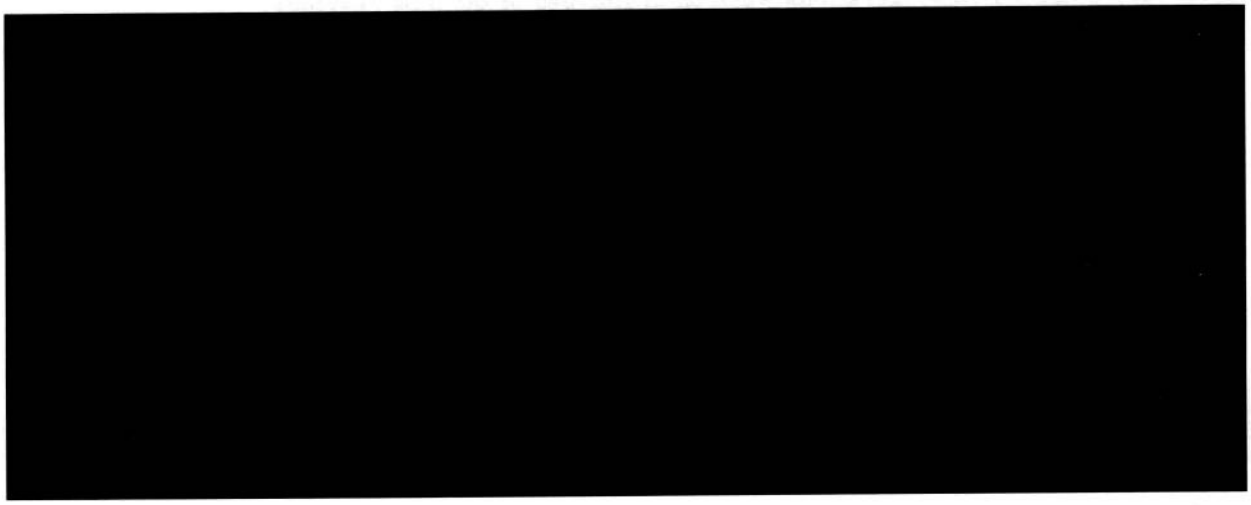
Part I

DESIGN AND OPERATIONAL CHARACTERISTICS OF THE S1W REACTORS

The reactors used in the S1W plant have been selected for the most detailed presentation of design procedures because of the extensive operating experience with these cores. The availability of operating data makes it possible to compare results of design calculations with operating results.

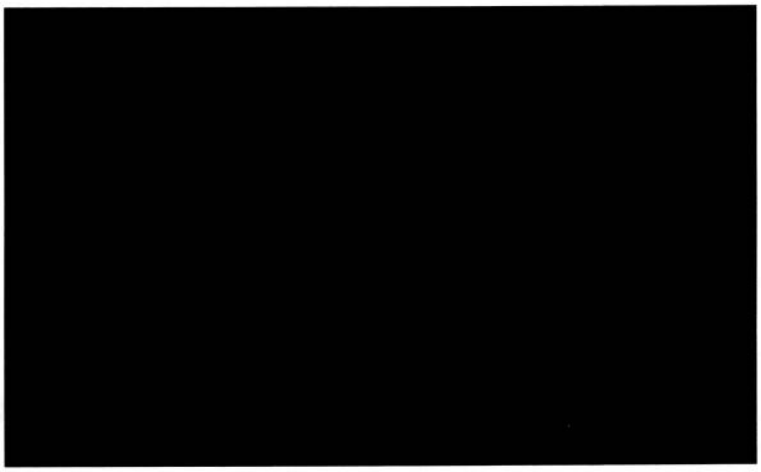
The S1W plant, originally designated the Submarine Thermal Reactor (STR), is the prototype plant for the submarine SSN 571, the NAUTILUS. The S1W plant, located at the Naval Reactor Facility in Idaho, was completed in 1953. The NAUTILUS plant is designated S2W and was completed in 1955. From the viewpoint of reactor design, these two plants are essentially identical.

There has been operating experience with four reactors in these two plants, three to the end of reactivity lifetime. Table I-1 gives the operational dates for these four reactor cores. In addition to these four, additional reactors for the S2W plant have been planned. Core 3B is a duplicate of Core 3. Cores 3C and 3D are designations of advanced type reactor cores for the S2W plant which are presently under development.



From a reactor physics point of view the mechanical designs of all of these cores are quite similar.

[redacted] There are, however, significant differences in the nuclear design between certain of these reactors. [redacted]



Since Cores 1 and 2 are essentially alike, except for a small difference in fuel loading, from a nuclear design point of view there are three different classes of cores which have been designed for the S1W and S2W plants.

It will be observed in the two chapters dealing with these reactors that the operational data obtained on the cores are not as complete or as satisfactory as would be desirable from a reactor physics point of view. There are two reasons for this. First, many tests were required with the S1W plant other than those concerned with reactor physics. These tests, many of an operational nature and of substantial duration, either prevented the performance of reactor physics tests or imposed such restrictions as to limit the amount and quality of physics data which could be taken. The second problem was simply the difficulty of making good reactor physics measurements in a complicated operating plant equipped primarily with operational type instruments even without interference from other tests. The operational data reported are those taken with the S1W plant. The data from the operating submarine are not available in sufficiently complete form to warrant analysis. This is not serious, however, since the S1W operations were on prototypes of the S2W reactors.

Over the ten-year period since the S1W reactor design was started, there have been major changes in the methods of nuclear design. In this interval there have been several generations of design methods developed and then made obsolete by the emergence of new techniques. Two factors have brought this about. First, the increase in understanding of reactor physics and, second, the advent of successively larger digital computers. The methods of nuclear design used to analyze the S1W reactors in the next two chapters are those which would be used at present if the designs were currently being developed. The presentation will be based on acceptance of the design as it exists and performance analysis of the design by current techniques.

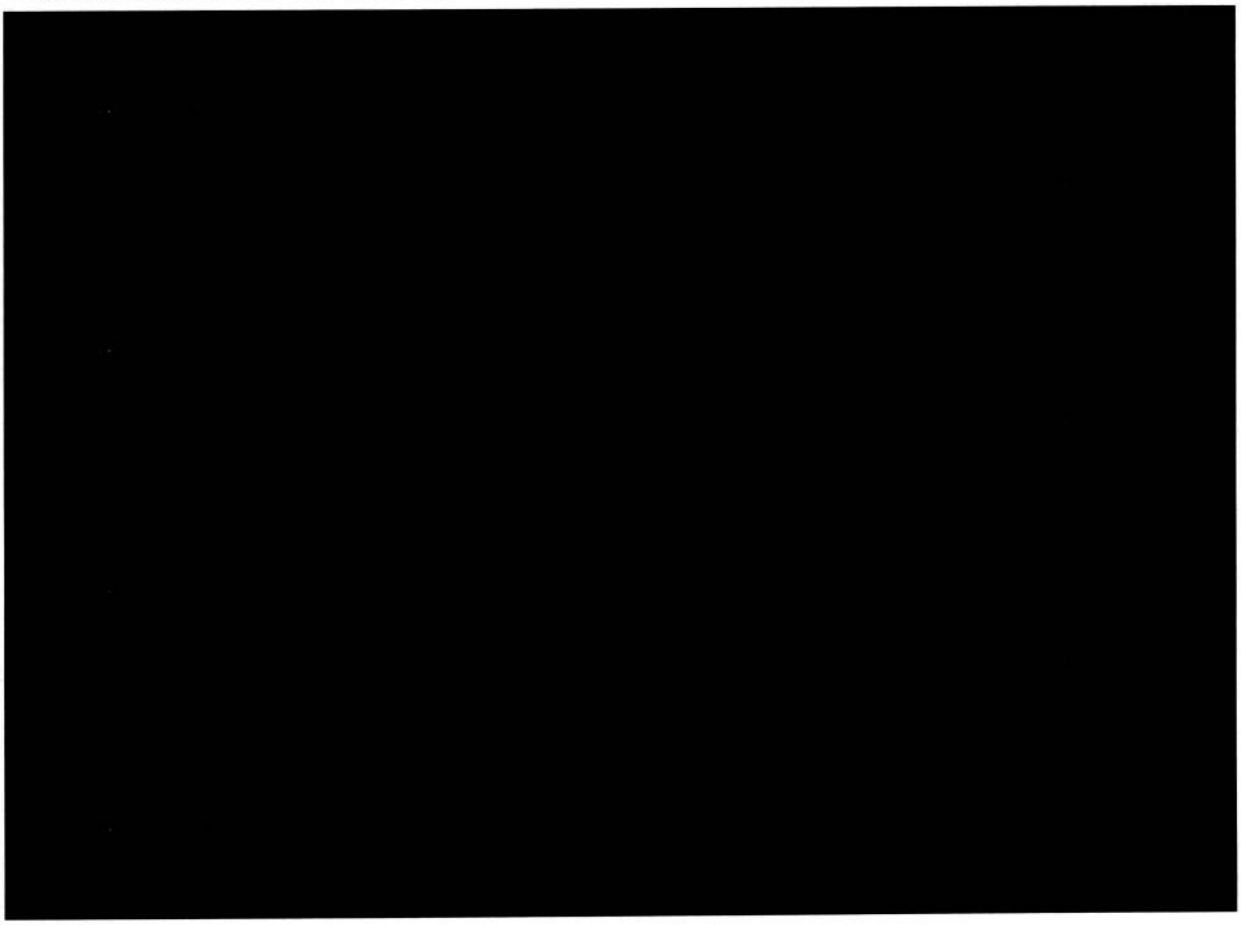
The division of the material between the two chapters is as follows: Chapter 2 deals with S1W Core 1 explicitly but covers Core 2 as well. [redacted] Operational results with the cores are included in the respective chapters.

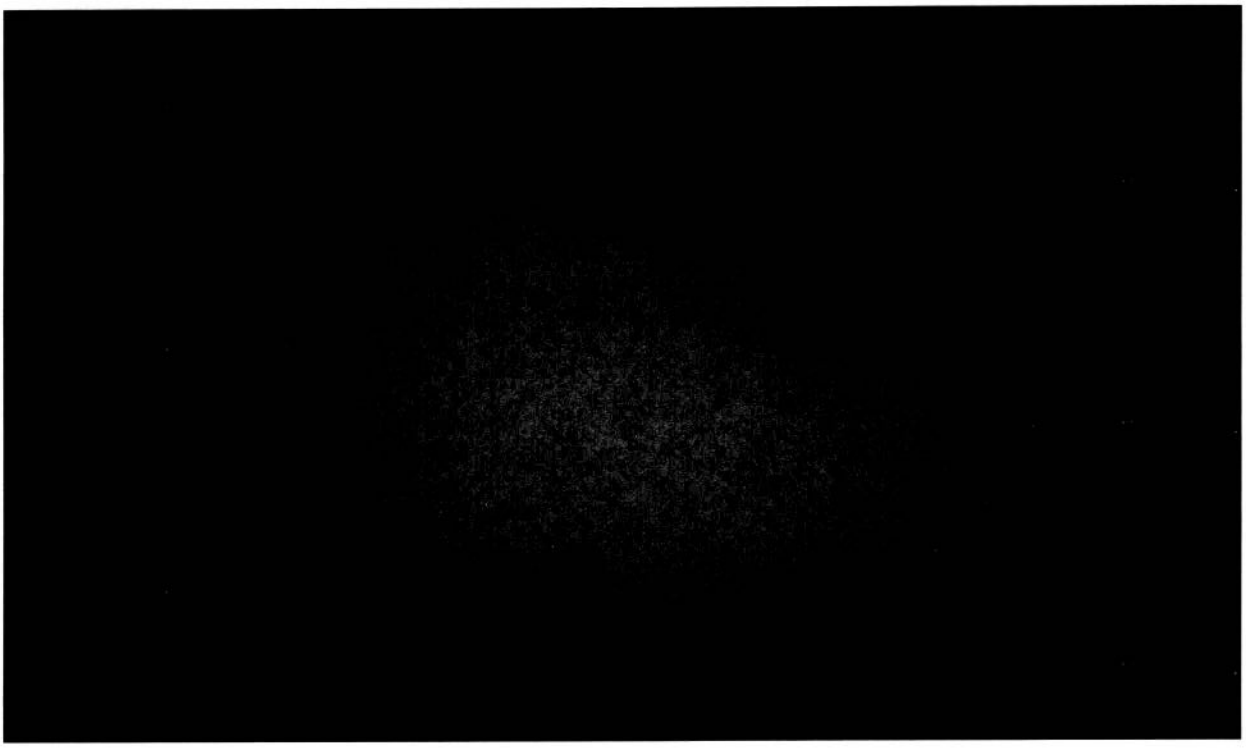
Pages 11 - 122 have been redacted in their entirety.

Part II

DESIGN STATUS OF CURRENT REACTORS

In this part of the handbook the nuclear design of the second generation of pressurized water reactors following the S1W and S2W, is reviewed. This group of reactors includes the PWR; two submarine reactors, the S3G, and SFR; and two reactors for surface vessel application, the A1W Cores 1 and 2. While the reactors in this group have certain elements of similarity, they differ from each other in many respects. This comes about not only because the applications are different and therefore different design features are required, but also because each reactor has been used as a vehicle for certain specific development objectives.





Chapter 4

DESIGN OF THE PWR

by R. T. Bayard and M. J. Galper

4.1 INTRODUCTION

The reactor concept chosen for the PWR is that of a seed and blanket reactor. The reactor consists of a small seed containing highly enriched uranium surrounded by a blanket containing natural uranium. The k_∞ of the seed is high, but that of the blanket is less than unity because it is fueled with natural uranium. The reactivity of the assembly is governed largely by the properties of the seed, and the blanket acts as a multiplying reflecting region. With this arrangement it is possible to extract a substantial amount of power from water moderated natural uranium. One important objective in this type of design has been the production of as great a fraction of the total power as possible in the natural uranium.

4.2 GENERAL PROPERTIES OF A SEED - BLANKET CORE

The principal advantage of the seed and blanket design, as compared with a single zone, slightly enriched design, is that a smaller quantity of thermally fissionable material is needed to maintain the reactor critical. The reason for this is that the fissionable material is concentrated in a region of greater importance for neutrons. An associated advantage of the seed and blanket design is that fewer control rods are needed to provide adequate reactivity control and shutdown than would be needed for the corresponding slightly enriched design. As the k_∞ of the blanket is always less than unity, no control is needed in this region. Control is associated only with the seed, which constitutes a small fraction of the total reactor volume. In a sense, it is necessary to provide one control rod for a given fraction of the total volume of reactor, since the rod poison cannot be felt at great distances from its position. Thus, it is necessary to provide more control rods of smaller individual worth in a slightly enriched reactor than for the corresponding seed and blanket reactor.

An additional advantage of the seed and blanket design is schedular. Since the blanket parameters do not markedly influence the reactivity or the temperature coefficient of the reactor, the design of the blanket can be selected and blanket fuel element manufacture can be started before the fuel loading of the seed has been established. In fact, different seeds containing different fuel loadings can be used with the same blanket. This flexibility permits the time between design and manufacture to be reduced. By way of comparison, in a single zone,

slightly enriched reactor, the loading, in terms of enrichment, must be set before fuel element fabrication can be started.

A problem with the seed and blanket design is the concentration of power in the seed and the relatively low power density in the outer blanket regions. The seed problem is met by use of plate construction which permits greater power density. The problem in the blanket is alleviated by limiting the coolant flow in the regions of lower power density.

The general features of the sharing of power between the seed and the blanket can be seen by considering the simplified expression for power sharing given below:[1]

$$\frac{\text{Blanket Power}}{\text{Seed Power}} = \frac{k^B \cdot \epsilon}{1 - k^B} \frac{k^S - 1}{k^S} \qquad \text{Eq. (4.1)}$$

where k^B and k^S are the infinite region multiplication constants of the blanket region and the seed region, respectively, and ϵ is the fast fission factor in the blanket.

In Eq. (4.1) the effect of neutron leakage out of the reactor is neglected. Nevertheless, it does indicate the general behavior of the power sharing in a seed and blanket reactor. As can be seen from Eq. (4.1), the fractional blanket power increases with increasing seed k^S. It also increases with increasing blanket k^B for k^B less than unity.

Over the operating lifetime of the reactor as equilibrium Pu^{239} concentration is established, k^B tends to approach a uniform value. This can be seen by ignoring for the moment the contributions of U^{235}, Pu^{240} and Pu^{241}, and fission product poisons. If the Pu^{239} concentration is low, then k^B is relatively low. According to the power sharing formula, the blanket power fraction is low. The fissioning rate in the seed and, hence, the fast neutron flux and leakage are high. Because of this large, fast neutron flux leakage into the blanket, the plutonium production rate is high and the plutonium concentration rises. As it rises, the blanket power fraction and k^B increase, and the number of excess neutrons from the seed diminishes. This simple picture is modified, of course, by the presence of U^{235}, and by the production and destruction of the higher plutonium isotopes, as well as by the build-up of the fission product poisons.

Various deductions can be drawn from Eq. (4.1) concerning the reactor design. Both k^B and k^S should be as large as possible. Considering over-all criticality, the requirement that k^S be large means that the seed should be small with high neutron leakage. This is confirmed by survey studies showing that large blanket power fractions can be obtained by using a small, high leakage seed. However, as k^S is made larger and the seed leakage increased by reducing the seed volume the seed power fraction does not decrease as rapidly as the seed volume. Thus, to obtain a high power fraction in the blanket, the power density in the seed must be high. In an actual design the limit on the blanket power fraction is set by the maximum power density which can be used in the seed.

To obtain high neutron leakage from the seed into the blanket, it is necessary to provide a high-buckling geometry in the seed. In general, this implies a seed with at least one of the principal dimensions rather small. A reactor design in which the dimensions of the seed are small in a two-dimensional sense (seed material in the form of long isolated cylinders) is possible; but this implies either a small single central seed, in which the power density would be high, or it suggests several seeds dispersed throughout the blanket. Such spiked seed designs have been studied. They appear to have peaking factors in the seed spikes which are

rather large. In the limit of a large number of small seeds of this type the reactor approaches a uniform, slightly enriched design.

For a practical reactor, designed to deliver a large output power, a geometry in which the seed is small in one dimension only appears to be most feasible. The annular seed of the PWR is such a design. In this design the seed leakage is governed by the thickness of the annulus. The volume of the annulus, which determines the power density, is adjusted by changing the radius of the annulus.

A detrimental effect of increasing the seed fuel loading is the increase in the flux peaking factor in the seed. As the seed loading is increased, the thermal absorption of the seed becomes larger than that of the blanket material, and a significant flux depression results at the center of the seed. Thus the over-all gain to be realized by increasing k^S through increased loading is limited by the flux peaking factor in the seed.

Since the reactivity of a seed and blanket reactor is determined principally by the seed and not by the blanket, the negative temperature coefficient of reactivity can be provided through the properties of the seed alone. For this reason the metal-to-water ratio of the seed must be fairly high. A water-starved condition is not needed in the blanket, and since a high k^B is desired, the water-to-uranium ratio can be chosen to give maximum k^B without regard to the temperature coefficient.

The PWR radial reflector is water. Since the reactor is quite large, the leakage from the reactor is fairly small and the importance of a reflector is reduced. The water reflector does tend to keep the flux in the outer regions of the blanket a little higher than it would be if the reflector were absent. This improves the peaking factor in the outer blanket region, but the gain is not large.

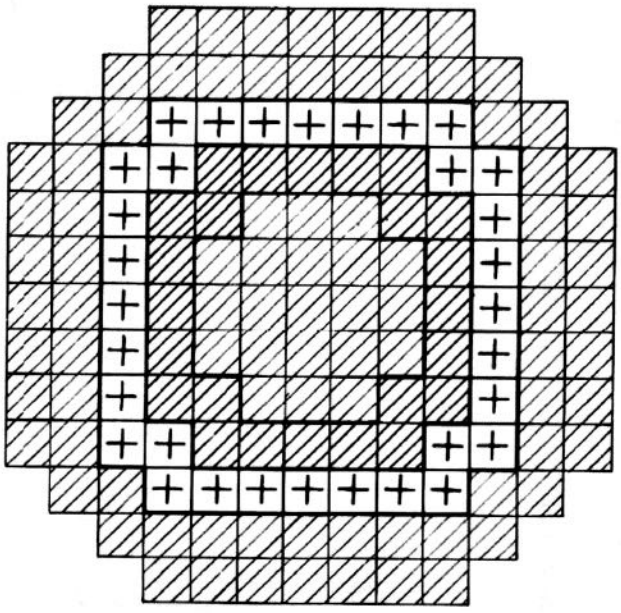
4.3 DESCRIPTION OF THE PWR REACTOR

(a) Arrangement of Materials. The over-all geometry of the PWR design involves an inner and outer blanket region surrounding an annular seed region. The core geometry is shown in Fig. 4.1. The seed is in the form of a modified square annulus with the corner seed units inset. The module, or center-to-center fuel assembly spacing, is 6 inches; the effective height of the fueled region in both seed and blanket is approximately 72 inches. The control rods are located in the seed units only.

Both the seed and blanket assemblies have external dimensions of 5-1/2 inches square. These fit into a pair of lattice plates which locate their bottom and top ends on the 6-inch center-to-center spacings.

There are 145 fuel assemblies in the design. Of these, 32 are seed assemblies and the remaining 113 are blanket assemblies.

(b) Seed Configuration. A detailed cross section view of a typical seed fuel assembly, called a cluster, is shown in Fig. 4.2. Each cluster consists of four subassemblies whose over-all dimensions are 2.50 inches by 2.50 inches. The subassemblies are made up of a series of parallel plates welded together at their edges to form an enclosed box. The fuel plates have a fuel section which is 0.039 inch thick by 2.050 inches wide by 70.75 inches long of uranium-235 alloyed in Zircaloy-2. The fuel alloy is a 6.5 w/o alloy of highly enriched (over 90 per cent) uranium-235 in Zircaloy. The fuel is clad with 0.015-inch-thick Zircaloy plates roll-bonded to the fuel alloy. In each subassembly the two outside plates have no fuel and are thicker than the fuel plates in order to withstand hydraulic forces. In the manufacturing



BLANKET ASSEMBLY

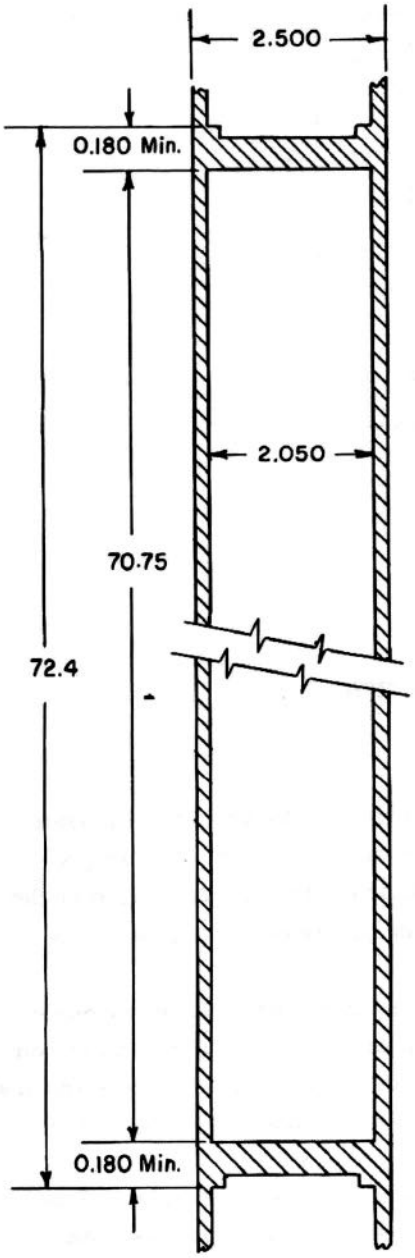
SEED FUEL ASSEMBLY WITH CONTROL ROD

Fig. 4.1 - PWR Core Geometry.

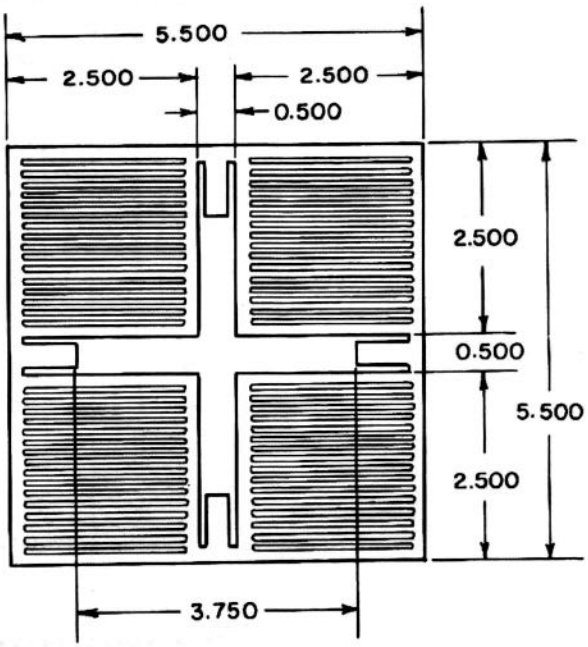
process the outside edges of the fuel plate are upset to give the plate cross section an I-beam shape; when the plates are stacked together, the I-beam sections are in contact. These provide the water channel coolant passages of 0.069 inch between plates. The flanged edges of the fuel plates when welded together constitute the remaining two side plates of the subassembly box.

The subassemblies are machine finished on the sides adjacent to the control rod channels. They are then assembled in groups of four into the cluster shown in Fig. 4.2. The control rod channel spacing is maintained by inserting Zircaloy spacer bars between the four subassemblies. These spacers are welded in place and constitute an integral part of the cluster. The outside of the welded cluster is machine finished to final dimensions.

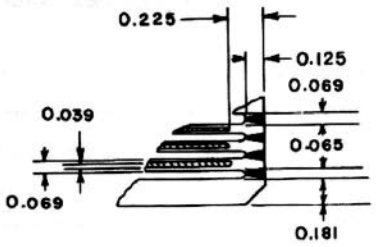
(c) Blanket Configuration. A blanket fuel bundle, shown in Fig. 4.3, consists of an array of Zircaloy-2 tubes each containing a stack of cylindrical uranium-oxide pellets. An individual fuel tube is shown in Fig. 4.4. The individual zirconium tubes are 10-1/4 inches long, including the two 1/2-inch long Zircaloy end caps. Each tube has 26 pellets; 0.358 inch long by 0.350 inch in diameter. The pellets are composed of natural uranium-dioxide sintered to approximately 93 per cent of theoretical density and ground to finished dimension. The nominal dimensions of the Zircaloy tube are 0.411 inch O.D. with a wall thickness of 0.022 inch. The Zircaloy end caps are welded to the tubes and then inserted into Zircaloy tube sheets containing a square array of 121 holes on 0.468 inch center-to-center spacing. The end caps are welded in place and the tube sheets are drilled in the spaces between rods to provide the water



CROSS-SECTION
OF
FUEL PLATE



CROSS-SECTION
OF
FUEL CLUSTER



ENLARGED VIEW
OF
CORNER OF SUBASSEMBLY

Fig. 4.2 - Seed Assembly.

passages. The assembled unit, called a bundle, consists of 120 fuel elements and has over-all dimensions of 10.25 inches high by 5.2 inches square. This is the unit shown in Fig. 4.3. One of the corner locations has no fuel element in order to provide a passageway for an instrumentation tube. Seven of these blanket bundles are stacked into a square 6-foot-long blanket assembly shell which is composed of 0.2-inch-thick Zircaloy plates welded into a box shape. The water passages of successive bundles in a blanket assembly are in register because the small lateral clearance between the bundles and the shell prevents misalignment of the holes.

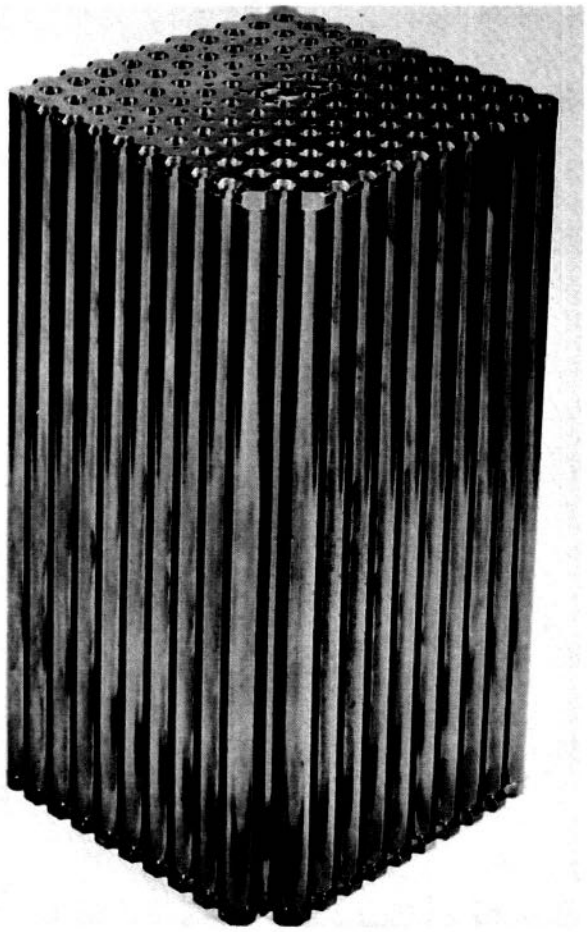


Fig. 4.3 - Blanket Fuel Bundle.

(d) Control Rods. The control rods are cruciform-shaped, and composed of hafnium of over 95 per cent purity. They are located in the 1/2-inch-thick cross-shaped hole in each seed cluster shown in Fig. 4.2. The span is 3-3/8 inches and the web thickness is 0.22 inch. The ends of the rods are rounded to ensure free passage of the rod in the channel. The control rods are driven by a screw-type mechanism and are scrammed by gravity alone.

4.4 OVER-ALL CORE CHARACTERISTICS - ONE DIMENSIONAL SURVEY STUDIES

The initial phase in the development of the PWR had as its goal the specification of a conceptual nuclear design. This served two purposes: First, it provided the basis for further development by mechanical and thermal designers of specific reference core parameters

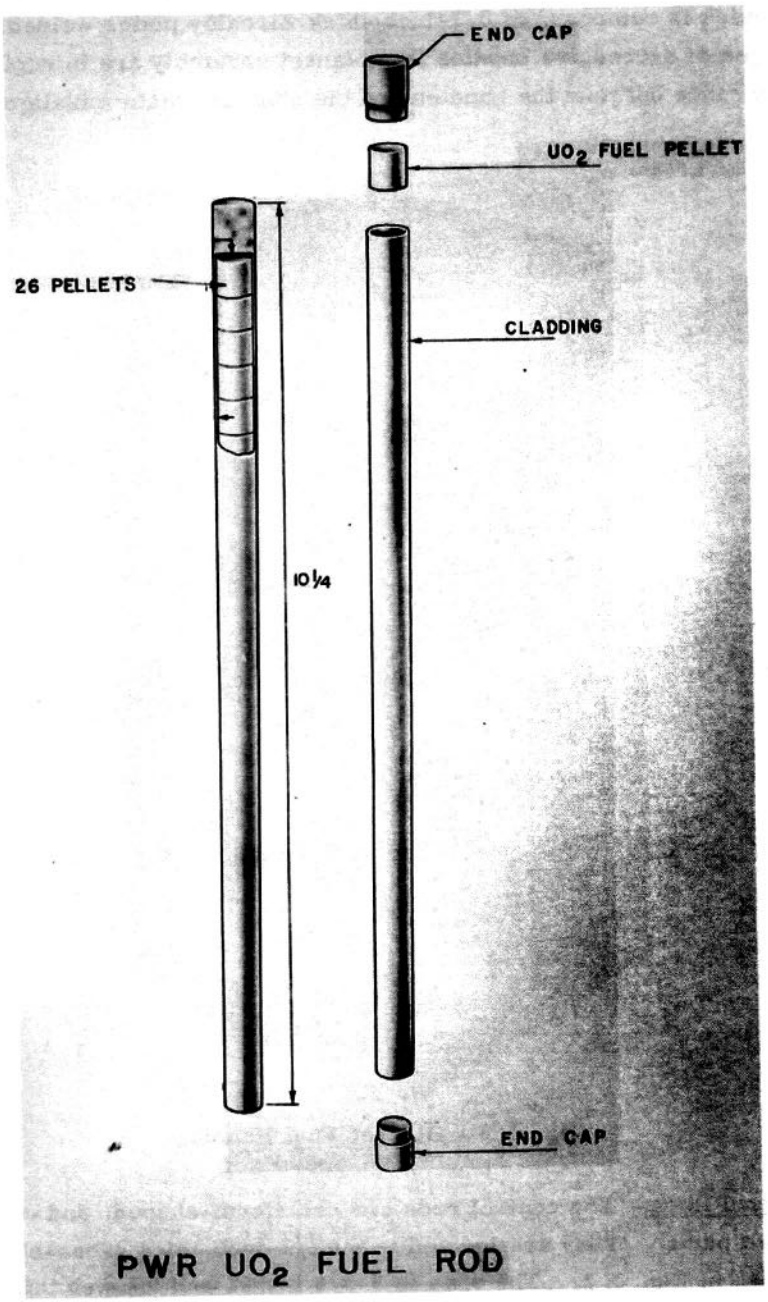


Fig. 4.4 - Blanket Fuel Tube.

compatible with the initial reactor physics information. Second, it established the requirements to be satisfied by a critical facility design, and allowed the design and construction of that facility to be initiated.

The process of developing physics information for a conceptual design requires a large number of reactor calculations. In the case of the PWR, the bulk of these preliminary survey calculations involved a simple, one-dimensional representation of the core. The core diameter and the radial positions of the interfaces between the seed and blanket regions were adjusted so that the area contained in each of the three fuel regions in the one-dimensional model was equal to that in the corresponding region of the actual two-dimensional core cross section. Figure 4.5 is a diagram of this model. Two-group diffusion theory was employed. The slow-group constants were found by averaging the thermal neutron cross sections over a Maxwellian energy distribution, and the fast-group constants were based on fitted microscopic cross section values. These fitted cross sections were derived from MUFT calculations (see Appendix E) using essentially the method described by Ombrellaro[3] (see also Volume I of this handbook). Resonance escape probability was computed by the method of Stein[2] (see also Volume I of this handbook).

The calculations were undertaken for the purpose of establishing interrelationships among the several quantities of interest. The criticality calculations were performed with a two-group, one-dimensional diffusion theory digital computer code AML-17 (see Appendix A). Lifetime depletion studies were made using the AML-67 code (see Appendix C). The quantities examined included: (1) The fraction of core power generated in blanket material for a reactor at criticality; (2) the effect of seed annulus thickness and location; (3) seed fuel loading for criticality; (4) fuel loading for a specified reactivity lifetime; (5) the effect of metal-to-water volume ratio in the seed; (6) the effect of water-to-uranium volume ratio in the blanket. In these calculations, the core was adjusted to criticality by assuming a uniformly distributed thermal poison cross section in the seed. Figures 4.6 through 4.10 summarize some of the results of this survey.

(a) Effect of Seed Thickness. It can be seen from Fig. 4.6 that a reduction in seed annulus thickness for a given mean radial seed location results in an increase in blanket power. However, as was mentioned in Sec. 4.2, it was found that the seed power density increases with decreasing seed volume. The engineering limitations on coolant flow and maximum metal surface temperature placed a limit on the allowable value for the seed power density. Thus, an upper limit was imposed on the attainable value for the blanket power fraction.

(b) Effect of Seed Location. A corresponding limitation was found when the seed annulus was moved inward toward the limit of a central seed core surrounded by blanket material. In this case also, the resulting increase in blanket power fraction was accompanied by an increase in seed power density. The seed position and annulus dimensions chosen for the PWR design thus represented a compromise between the competing effects of blanket power fraction and seed power density.

(c) Effect of Water-to-Uranium Volume Ratio in Blanket Material. In order to minimize the amount of highly enriched uranium fuel required to provide a given core lifetime, a high value of blanket power fraction is desirable. As mentioned previously, a parameter which affects this power fraction is the multiplication constant, k^B, of the blanket material. Based on the experimental evidence cited in Chap. 13, this is optimized in the PWR blanket by the choice of a water-to-uranium volume ratio of between 3:1 and 4:1. (This ratio is specified as

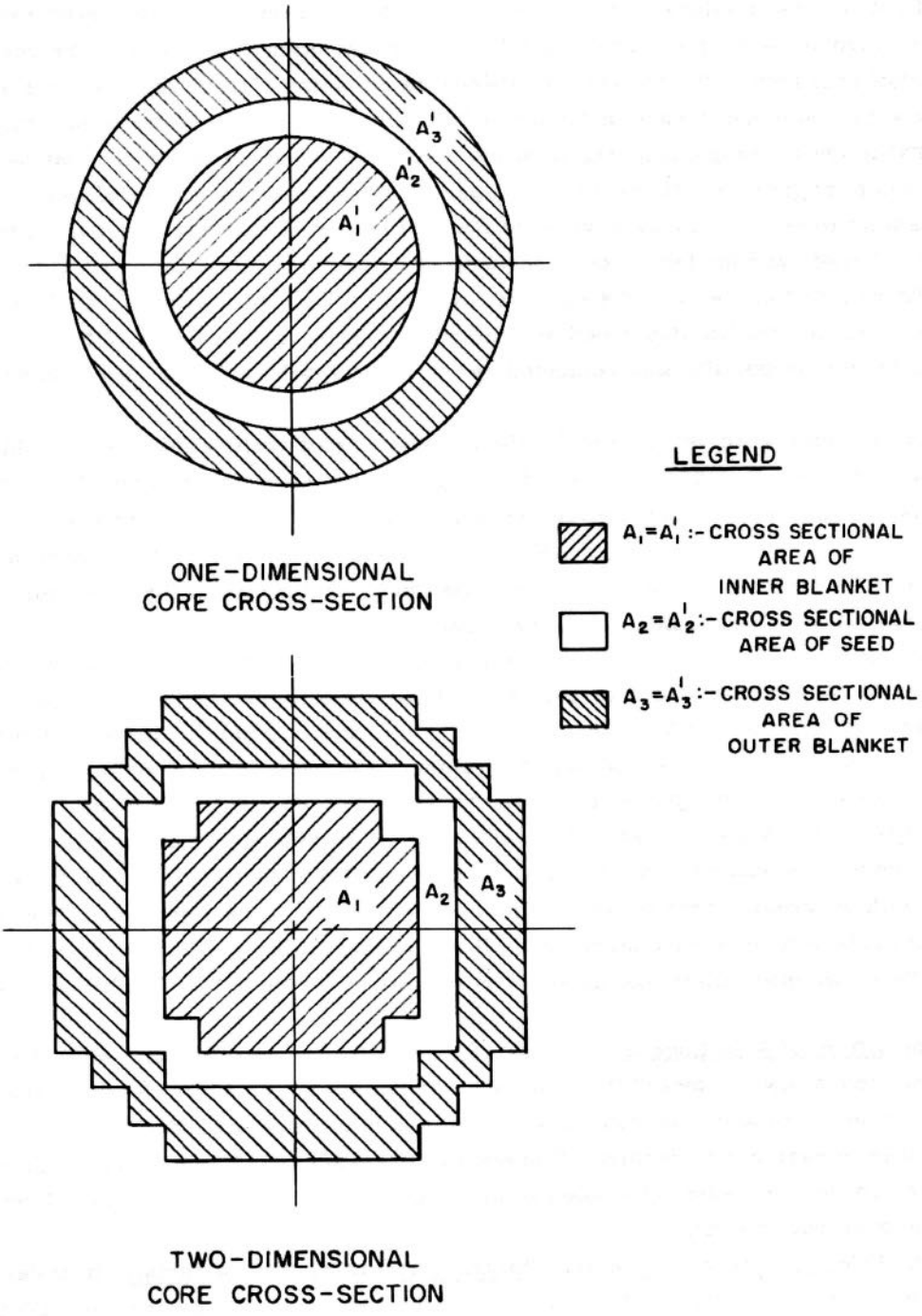


Fig. 4.5 - Relation of One- and Two-Dimensional Cross Sections.

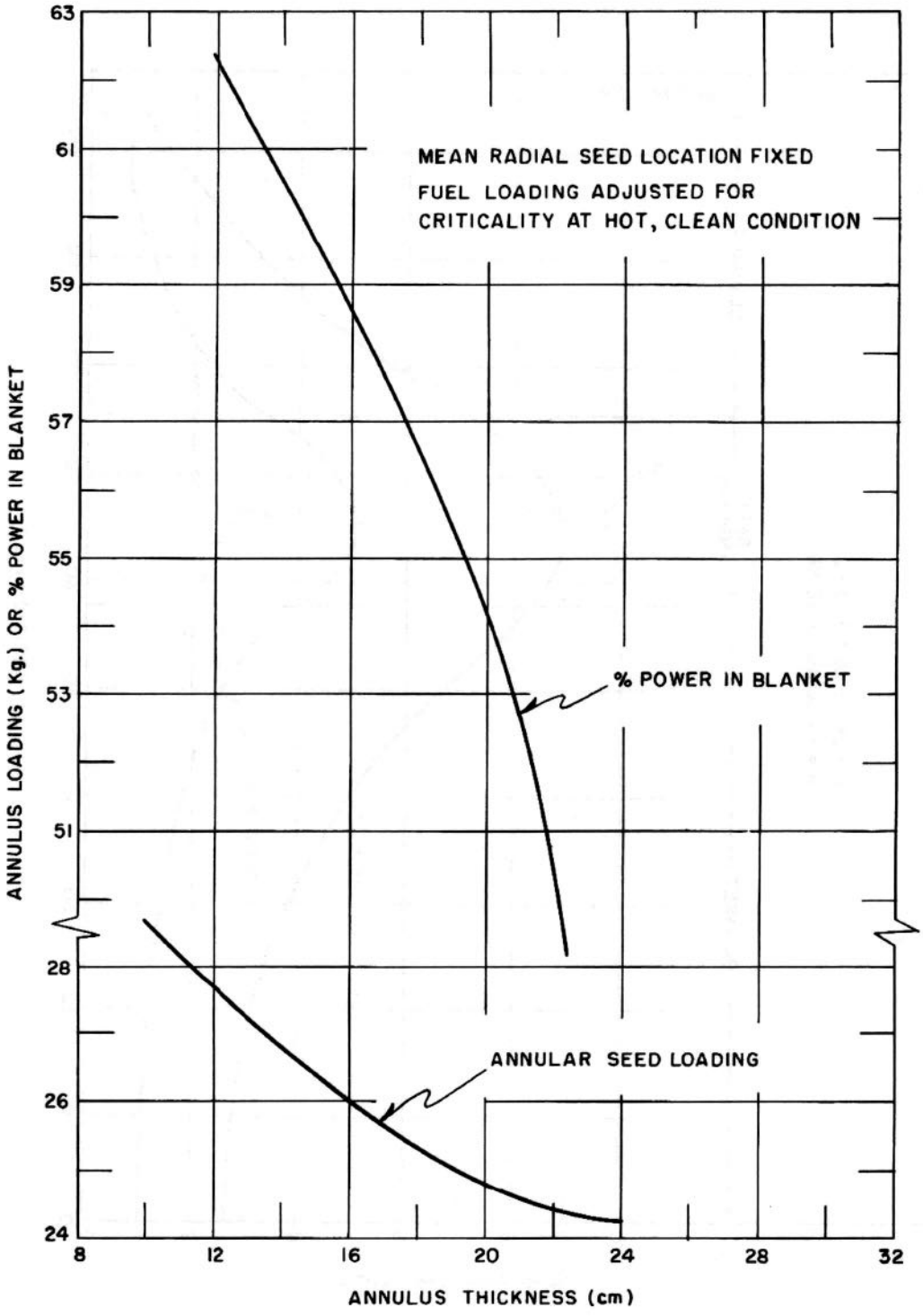


Fig. 4.6 - Seed Loading and Blanket Power versus Annular Thickness

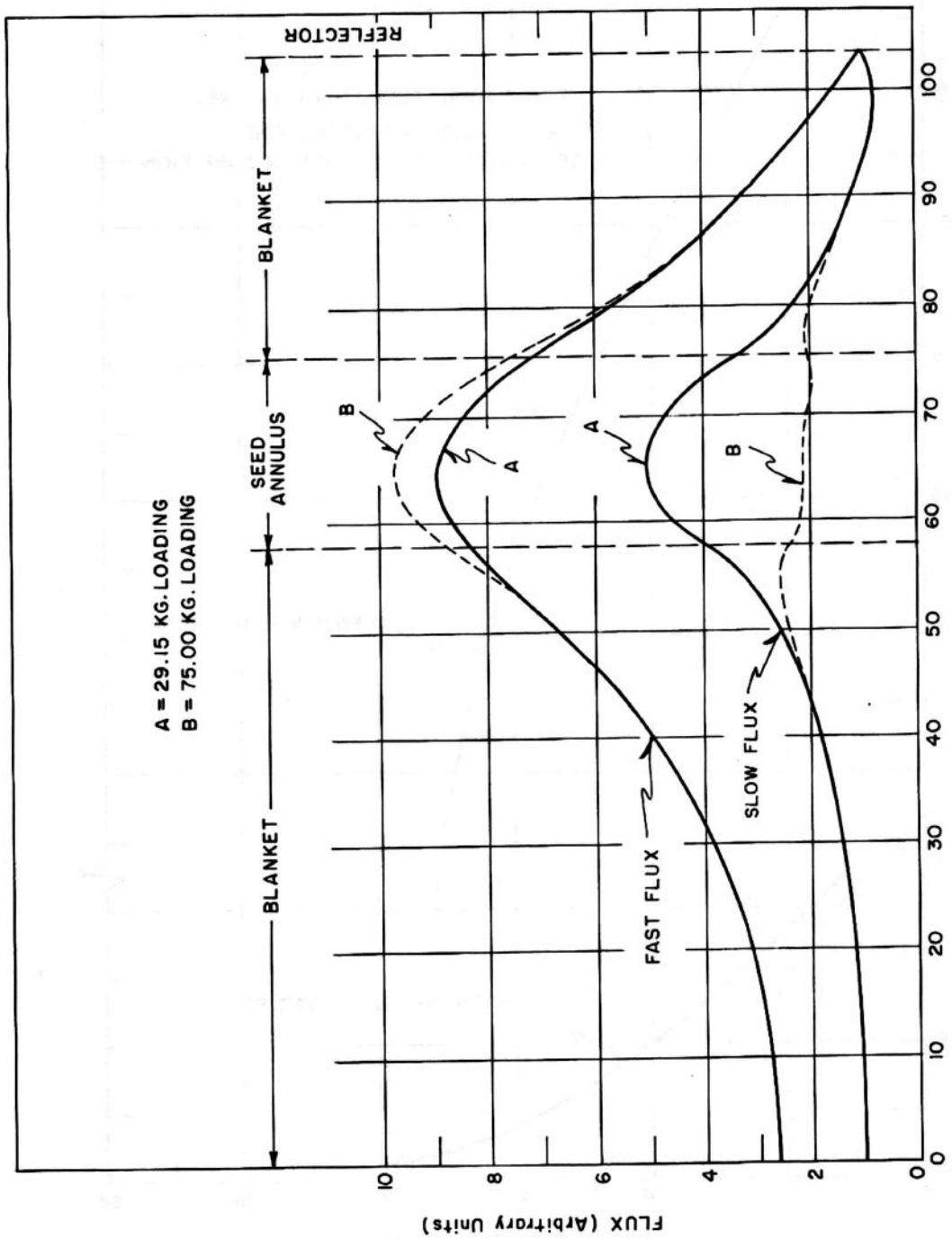


Fig. 4.7 - Effect of Seed Fuel Loading on Flux Shapes.

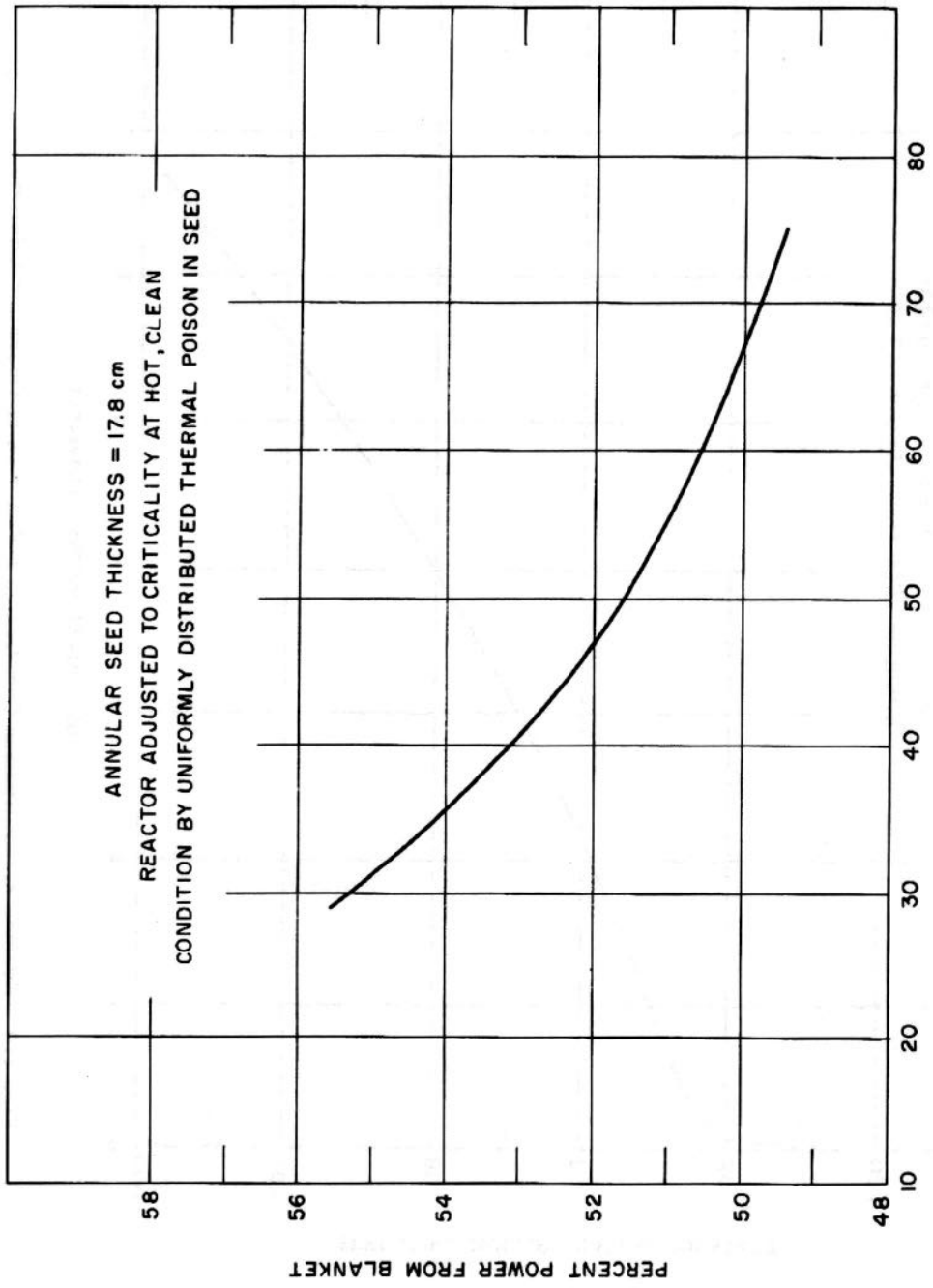


Fig. 4.8 - Blanket Power as a Function of Seed Fuel Loading.

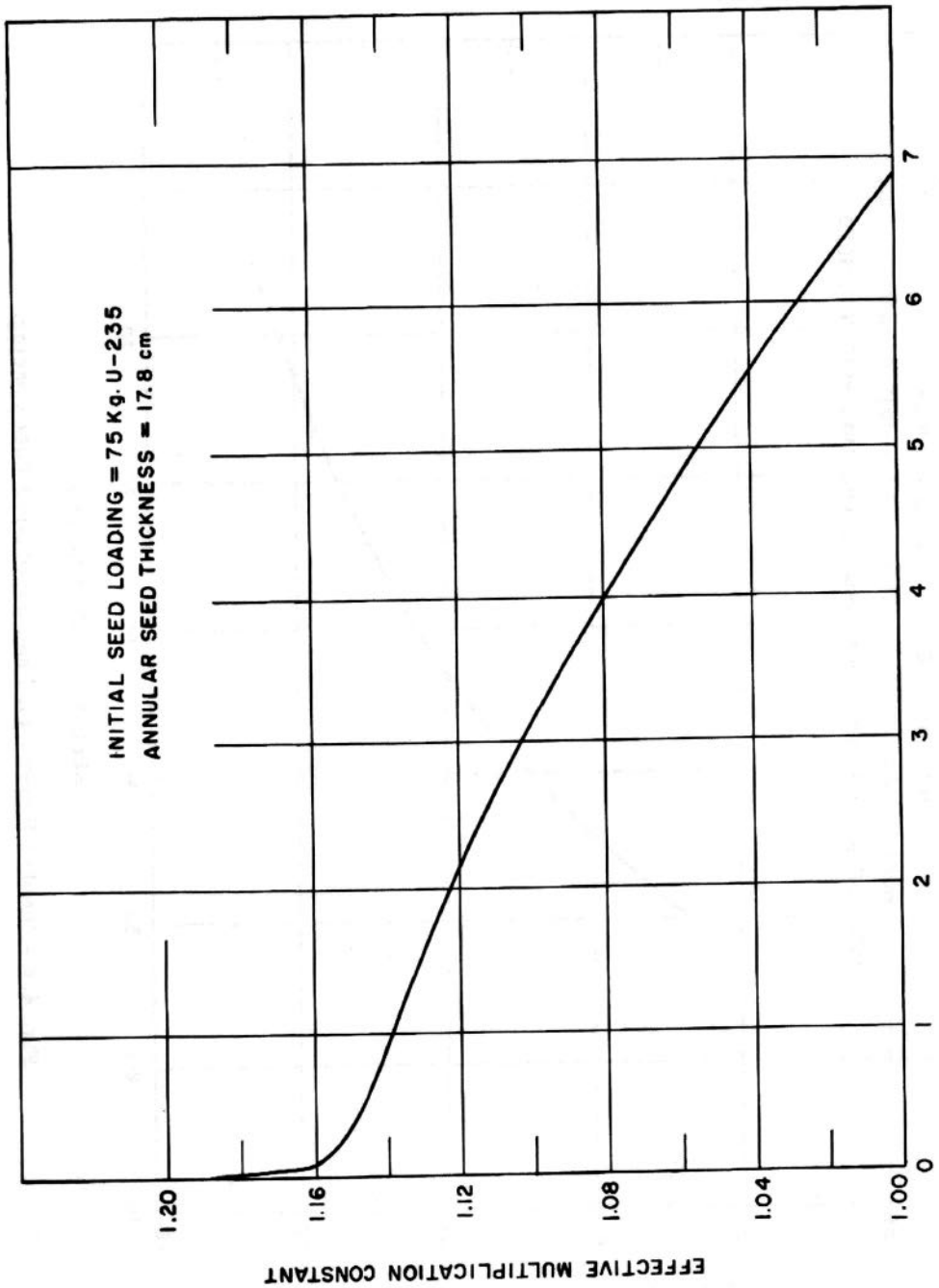


Fig. 4.9 - Reactor Multiplication Constant as a Function of Operating Lifetime.

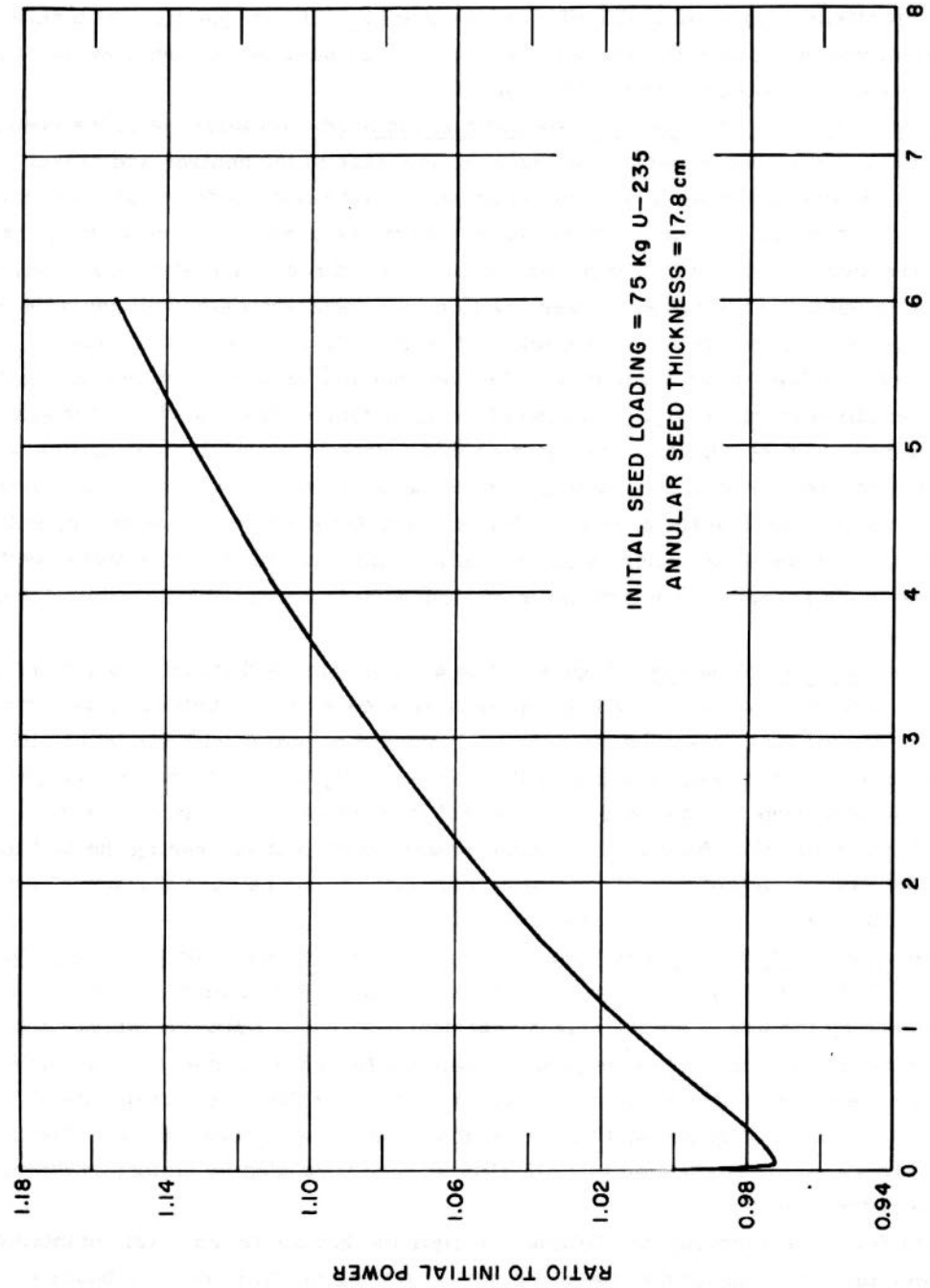


Fig. 4.10 - Variation in Blanket Power Fraction with Reactor Lifetime.

the water-to-uranium volume ratio with the uranium considered as metal.) In a single zone, slightly enriched core, an upper limit on the water-to-uranium volume ratio is set by the temperature coefficient of reactivity (essentially $\partial k_\infty/\partial T$) which must have a negative value to provide stability. In the PWR seed and blanket design, calculations have shown that the blanket k_∞, although important in determining power distribution, had only a second order effect on core reactivity compared to the effect of the seed k_∞. Consequently, the blanket water-to-uranium volume ratio was chosen to have a value as near the optimum as thermal and mechanical design considerations would permit.

(d) Effect of Metal-to-Water Volume Ratio in the Seed. An increase in the metal-to-water volume ratio of the seed material results in an increase in the neutron age of that material. The apparent thickness of the seed annulus for fast neutrons decreases with increasing neutron age. Therefore, the fast neutron leakage from the seed, and consequently the blanket power fraction, is higher for larger values of the seed region metal-to-water volume ratio. In order to optimize the blanket power fraction, the metal-to-water volume ratio for the seed was chosen to be as high as mechanical and thermal limitations would allow.

The choice of metal-to-water volume ratio for the seed has an important bearing on the temperature coefficient of reactivity. Because of the high neutron leakage characteristic of the thin annular seed, the reactivity contributed by this region is particularly sensitive to variations in neutron age. A change in moderator temperature alters reactivity because it affects both the neutron age and the seed k_∞. The former, however, is the more important effect. A high value of metal-to-water ratio results in a more negative temperature coefficient of reactivity and tends to make the reactivity more sensitive to small variations in mechanical dimensions.

(e) Effects of Seed Loading. Figure 4.7 is a radial plot of thermal-group and fast-group flux distributions for a typical cylindrical annular seed with a cylindrical inner and outer blanket. The flux plots describe the reactor in the hot condition with two values of seed loading. At each loading the reactor was adjusted to criticality by a uniform thermal poison in the seed. A justification for this method of control is discussed below in Sec. 4.5.

Figure 4.8 describes the effect on the blanket power fraction of increasing the fuel loading in the seed to provide for depletion with operating lifetime. It can be seen that one effect of fuel depletion is to increase the blanket power fraction.

(f) Reactivity Lifetime Calculations - Gross Changes of Power Distribution with Life. The effects of U^{235} fuel depletion and plutonium build-up on the distribution of power in the core was examined by the use of a one-dimensional depletion code, AML-67, which calculated the timewise variation of fuel and fission product isotopes for a two-group, one-dimensional model of the core (see Appendix C). At each step in a series of discrete increments of time, the effective multiplication constant and the power distribution were recalculated. The latter was computed for a reactor readjusted to criticality at each time step by uniformly distributed thermal poison in the seed.

It was assumed in performing the lifetime calculations that the reactor was maintained at a constant power level throughout the life of the core. During the first 40 to 50 hours of operation, the initially clean core was markedly affected by the build-up of the xenon fission product poison. Figures 4.9 and 4.10, which describe the variations in effective multiplication constant (k_{eff}) and blanket power fraction with time, demonstrate the effect of xenon build-up

on these quantities. After approximately 50 hours of operation, the xenon concentration attained an equilibrium value which then varied only slightly throughout core life., Thus, for times in excess of 50 hours, the curves describe the effect on k_{eff} and the blanket power fraction of fuel depletion and plutonium build-up. The increase in blanket power with time, as described in Fig. 4.10, includes the effects of fuel depletion in the seed and the build-up of Pu^{239} in the blanket.

These one-dimensional reactivity lifetime calculations were used to estimate the fuel loading in the seed necessary for core life. Since the seed power fraction does not change greatly with lifetime, the fuel loading can be estimated to within 10 per cent by adding to the critical loading sufficient fuel to provide for the depletion which would result if the seed power fraction remained constant at its initial value throughout core life.

4.5 OVER-ALL CORE CHARACTERISTICS - TWO-DIMENSIONAL STUDIES

To utilize the results of physics investigations on the distribution of power in the reactor, the reactor thermal designer requires more information than can be obtained solely from a simple one-dimensional representation of the core. Needed information includes the magnitude and location of the power peaks in the various orificing regions of the core, the fraction of the total core power generated in the various regions, and the effects of different control rod configurations.

To provide such information, reactor calculations were performed by means of the CUTHILL code and the QED and PDQ codes. These are two-dimensional, few-group, diffusion theory codes for use with digital computers (see Appendix B). The limitations of the digital computer allowed the reactor to be described only on the basis of quadrant symmetry and without any description of the local heterogeneity of the core. The presence of a control rod in a seed cluster, for example, was represented by a uniform poison in that cluster. Figure 4.11 is a contour map of the thermal group flux from a typical two-dimensional calculation for a hot clean core in which it was assumed that control rods were fully inserted in the four inset corner seed clusters (the shaded block) and all other control rods were fully withdrawn. This calculation was made for the radial plane, i.e., the plane normal to the control rod axis. Similar two-dimensional calculations were performed for other control rod configurations, and a description of the distribution of power in each subregion of the core was thus obtained.

Two-dimensional calculations, such as that of Fig. 4.11, do not, in general, represent the reactor in its critical state. If it were possible to describe the reactor by a three-dimensional calculation, a supercritical configuration would be adjusted to criticality by the insertion of additional control rods downward into the core. A two-dimensional calculation for a seed and blanket reactor does not allow this mode of control to be described. The fact that the two-dimensional calculation does not represent a critical condition affects the blanket power fraction obtained from the two-dimensional calculation. This problem was investigated by comparing a series of one- and two-dimensional calculations.

In Fig. 4.12, the results of a series of two-dimensional calculations for various rod configurations are summarized in terms of the blanket power fraction corresponding to a particular value of k_{eff}. In addition, the results of a number of one-dimensional calculations are shown in which the k_{eff} of the core was varied by adjusting the amount of uniform thermal poison in the seed. The results shown in Fig. 4.12 indicate, first, that the blanket power fraction is a function of the excess reactivity of the reactor, and second, the one-dimensional blanket power

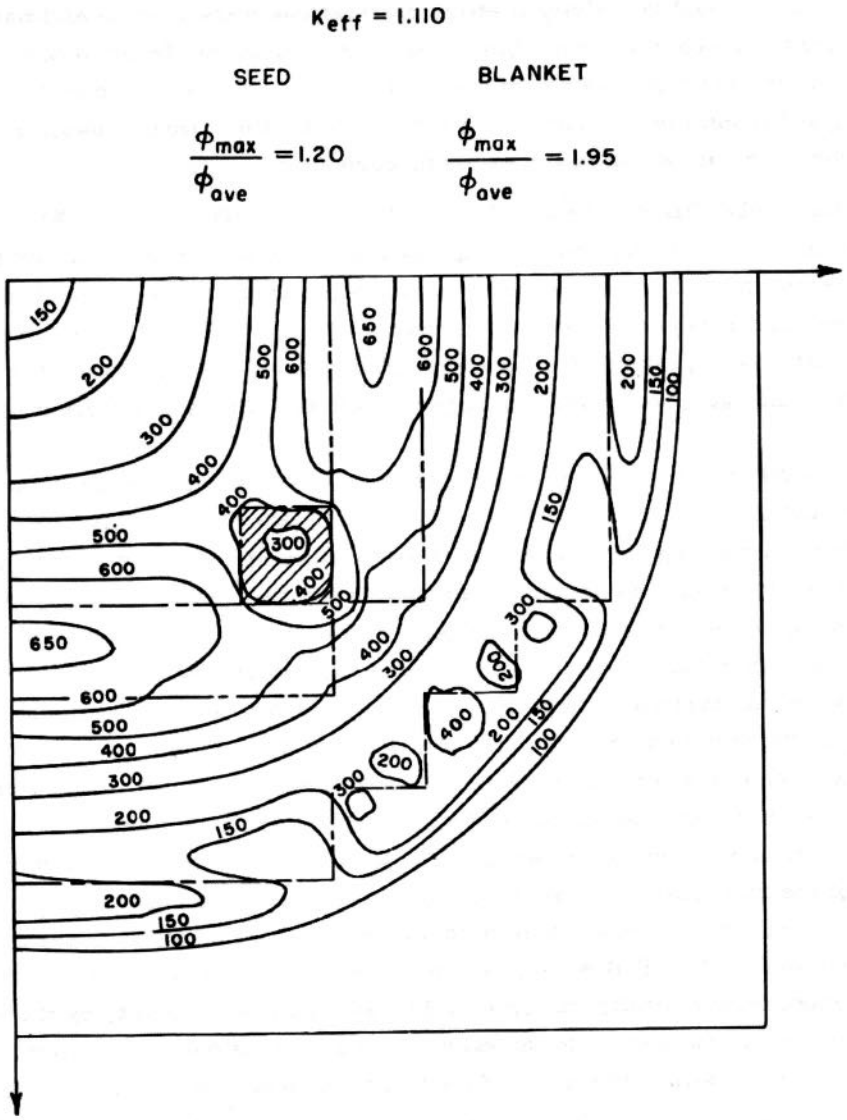


Fig. 4.11 - Two-dimensional Calculation of Thermal Neutron Flux in a Radial Plane. Reference Core Design with 4 Control Rods Inserted. (Arbitrary Normalization).

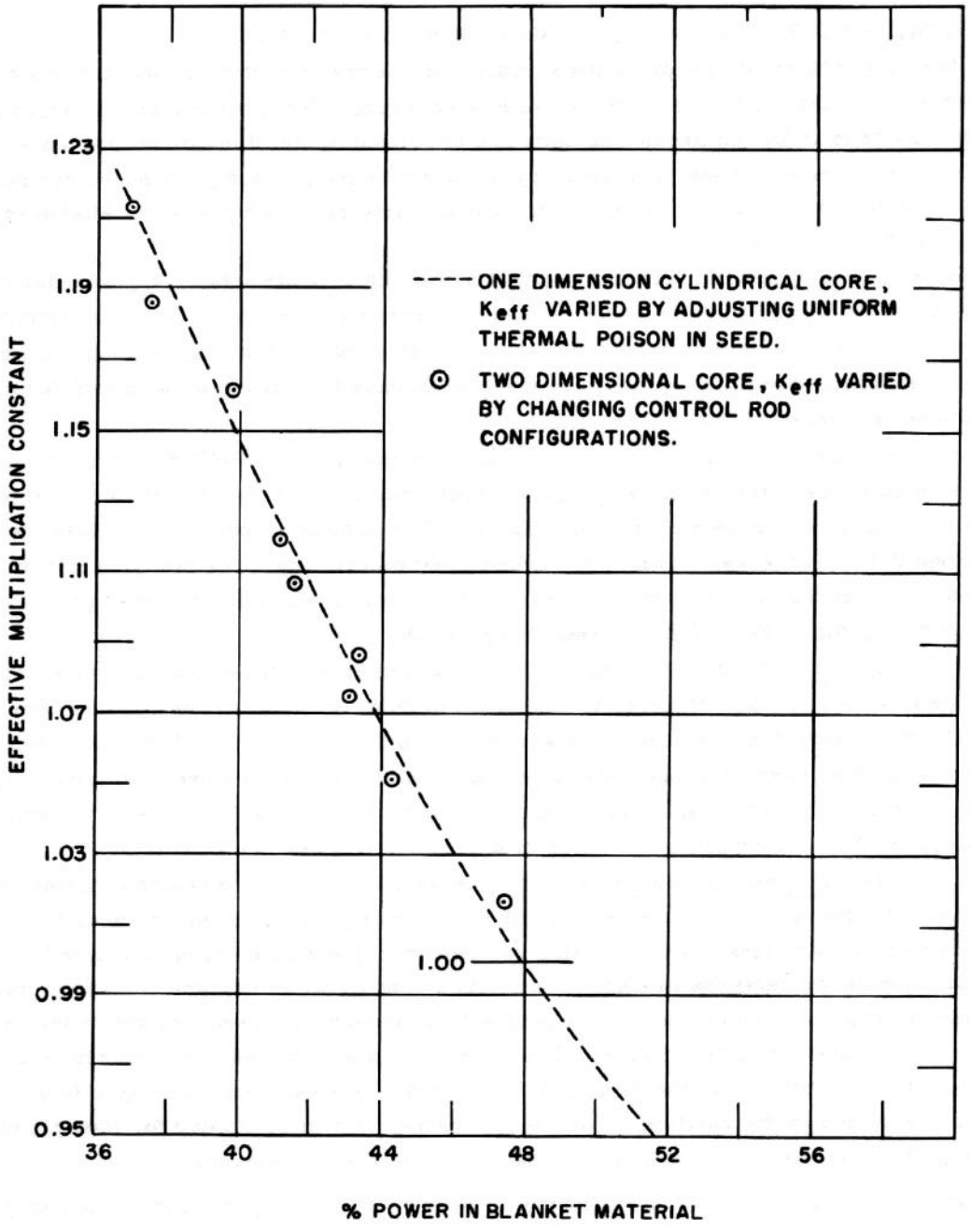


Fig. 4.12 - Relation Between Blanket Power Fraction and Effective Multiplication Constant of the Reactor.

fraction compares quite well with the corresponding value obtained from the two-dimensional calculation. On the basis of these results, the two-dimensional calculations are used to obtain peak-to-average flux ratios in various regions of the core; however, one-dimensional calculations at criticality are relied upon to establish the blanket power fraction.

4.6 DETAILED CORE CHARACTERISTICS - GROUP CONSTANTS

In the early stages of design analysis, when the interrelationships among the basic core parameters are examined in order to arrive at a conceptual design, the fact that oversimplified representations for the group constants are employed is not of major consequence. Any refinement of the calculational techniques by a more elaborate description of the neutron energy distribution will not, in general, affect qualitative relationships of the kind described in the preceding sections.

However, it is known that reactor calculations which provide data on k_{eff} and flux distributions are significantly affected by the number of energy groups used and by the methods employed for determining the group constants. Techniques which are quite adequate for a core geometry less complicated than that of the PWR have been found to be unsatisfactory for calculations of a seed and blanket design.

Comprehensive analysis of the experimental data taken on the PWR Flexible Assembly has been used to test various analytical procedures and to select the one giving the most consistent agreement with experimental information. The procedure selected involves the use of four-group diffusion theory, in which the group constants for the three fast groups are obtained by use of the MUFT codes (see Appendix E), and in which the thermal group constants are provided by the SOFOCATE code (see Appendix D).

(a) The Fast Group Constants. The few-group theory of neutron diffusion, which forms the basis for the MUFT code, is described in detail in Volume I of this handbook. The isotopic number densities that comprise a portion of the input for the MUFT calculation are obtained by volume averaging the material concentrations in the core over the gross regions of seed, blanket, and reflector. The input parameter, B, required by the MUFT code, has been found to affect only slightly the value of k_{eff} given by a reactor calculation.

(b) The Thermal Group Constants. It is well known that the neutron thermal energy spectrum is hardened, i.e., weighted toward higher energies, by the strong energy dependence of most thermal absorbers. The resulting neutron energy distribution is calculated by the SOFOCATE code for neutrons in equilibrium with a homogeneous mixture containing monatomic hydrogen undergoing thermal motion. The effects of absorbers having any arbitrary energy dependence are described by the code. The code calculates macroscopic cross sections for any mixture of isotopic constituents with water with the cross sections appropriately weighted by the neutron energy distribution. The energy distribution is calculated for isotopic compositions identical with those used in obtaining the fast-group constants.

4.7 DETAILED CORE CHARACTERISTICS - TREATMENT OF LOCAL HETEROGENEITIES

The information obtained from the studies described in Secs. 4.4 and 4.5 above provides most of the data needed for the development of the conceptual design. However, a number of important factors must be analyzed. In particular, the effects of local geometric heterogeneity must be examined because (1) these to a large extent determine the location and magnitude of temperature hot spots; and (2) the heterogeneities affect the reactivity. Both problems arise because of the relatively thick water channels separating the fuel assemblies. The

thermal flux peak in the water channel increases the fission rate in the fuel adjacent to the channel. Moreover, the thermal flux peak raises the average thermal flux in the water above that in the fuel and, as a consequence, reduces the effective multiplication constant of the region. The problem of power distribution is treated in this section, while the reactivity effects are discussed in the following section.

Water channels are present in the PWR design for two reasons. First, there is a water channel, approximately one-half inch thick, around each blanket fuel bundle and each seed cluster to permit removal of individual fuel assemblies from the core structure. Second, within each seed cluster there is a cruciform shaped water channel for the control rod (see Fig. 4.2). This channel is one-half inch thick.

The effects on the thermal group flux distribution of local variations in core structure were examined by assuming that these variations could be treated independently from the somewhat smoother variations in flux caused by the finite size of the core and from the substantially different nuclear properties of the materials for the seed and blanket regions. This separation of gross and local effects was accomplished by examining the thermal flux distribution in a unit cell of an infinite medium, where the cell consisted of the fuel assembly and its surrounding water annulus. One-group diffusion theory was employed to describe the thermal flux distribution. The neutron source was assumed to be constant in each subregion of the cell structure and proportional to the hydrogen density. The boundary conditions were provided by the assumption that there was no net flow of neutrons across the cell boundaries. Thermal group constants were obtained by use of the SOFOCATE code. One- and two-dimensional calculations were made.

(a) <u>One-Dimensional Calculations.</u> The water channel peaking factors of principal importance in calculations of core thermal performance can be described adequately only by two-dimensional cell calculations which take into account the effects of the right angle intersection between two water channels. One-dimensional calculations, however, are of considerable value in establishing the degree of heterogeneity required in a two-dimensional calculation to describe adequately the flux variations in the fuel assembly.

A series of one-dimensional calculations were made in which the geometric detail explicitly represented was varied. The calculation represented a traverse normal to the fuel plates of a seed subassembly. In one calculation only two regions were represented: the water channel and zirconium end plate; and the fuel plates and coolant water channels. In the second calculation four regions were represented: the water channel; the zirconium end plate; the first water coolant channel; and the homogenized fuel and remainder of the coolant channels. In the third calculation ten regions were represented: the water channel, zirconium end plate and first coolant channel; the next four coolant channels; the first four fuel plates; the remainder of the fuel and coolant channels homogenized. Figure 4.13 shows the result of this study.

Figure 4.13 indicates that the first calculation, namely separate homogenization of the non-fuel bearing region and the fuel bearing region of the cell, is satisfactory for calculating the thermal flux distribution. The peak-to-average flux is overestimated somewhat by this method of homogenization, but this has been accepted as a factor of conservatism inherent in the cell calculations for water channel flux peaking factors.

A flux peaking factor of interest in the thermal design is one found in an axial traverse of a blanket assembly. This factor can be computed directly by a one-dimensional calculation.

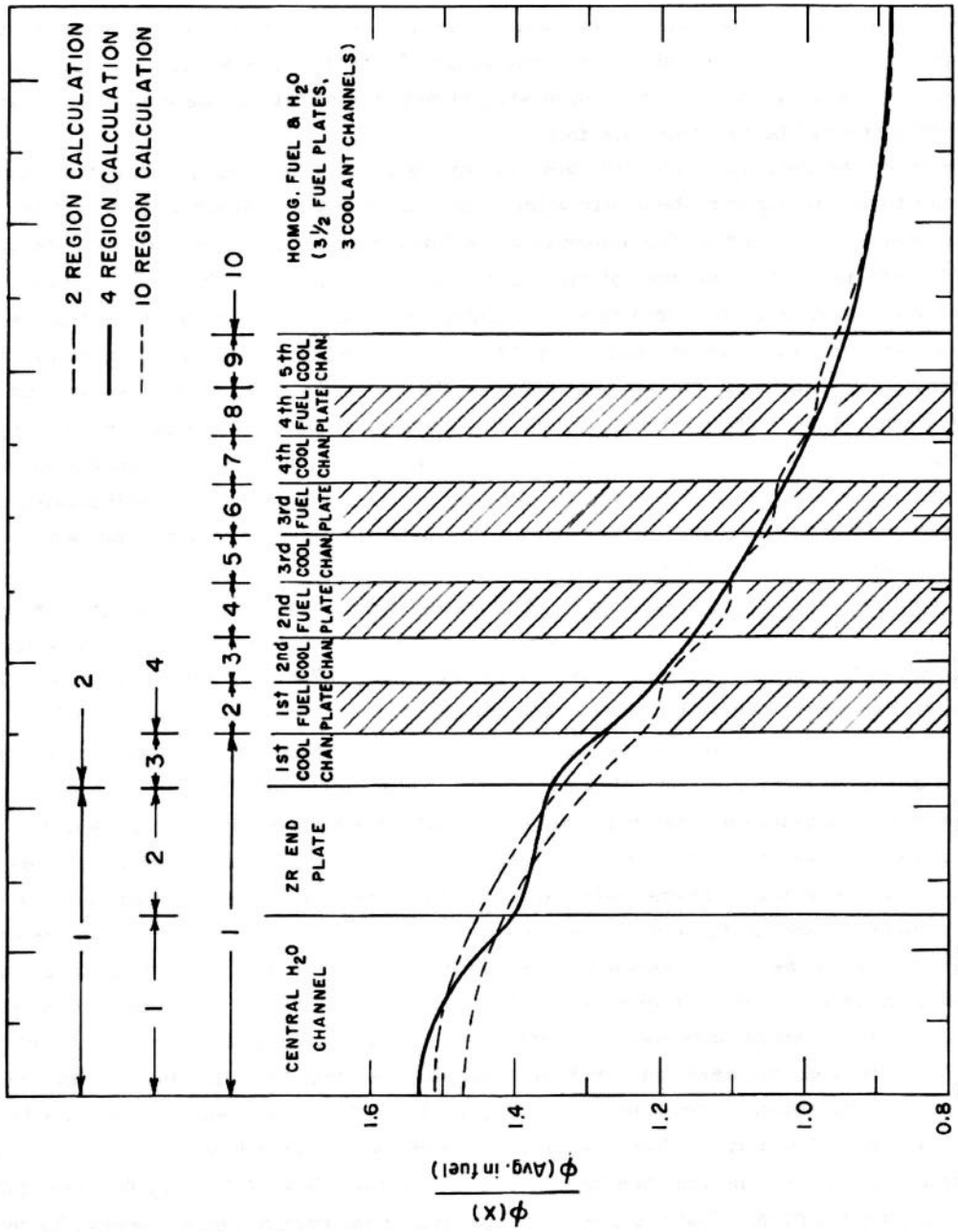


Fig. 4.13 - One-Dimensional Cell Thermal Flux Distribution in Seed Subassembly.

It results from the separation of the UO_2 fuel in the blanket fuel elements into 9-1/4-inch segments which are separated by one-inch-thick layers of Zircaloy and water that comprise the blanket bundle tube sheet regions (see Sec. 4.3c). The results of this calculation are illustrated in Fig. 4.14.

(b) Two-Dimensional Calculations. Based on the results of the one-dimensional analysis, two-dimensional cell calculations were performed to obtain the thermal flux distributions in typical seed and blanket assemblies. The fuel bearing region was assumed to be homogeneous in composition and surrounded by a homogeneous non-fuel bearing region which included the large water channel and the metal wall structure. Figures 4.15 and 4.16 describe the thermal flux contours provided by cell calculations for a seed subassembly and a blanket fuel assembly, respectively.

4.8 DETAILED CORE CHARACTERISTICS - REACTIVITY AND POWER DISTRIBUTIONS

(a) Flux-Weighting of Thermal Constants. Because of the spatial variations in thermal neutron flux resulting from the heterogeneous structure of the core, the average thermal flux in the water moderator is higher than that in the fuel. In other words, the relative thermal absorption in fuel and non-fuel materials is equivalent to that in a homogeneous seed-blanket reactor having a reduced quantity of fuel.

This effective reduction in neutron absorptions by the fuel would be calculated directly by a two-dimensional reactor code which could describe the detailed core structure. However, the two-dimensional code and computers presently available are incapable of allowing explicit description of core heterogeneity for a system as large as the PWR.

In order to take into account the effects of heterogeneity on criticality and gross power distribution, a technique utilizing the one-group, two-dimensional cell calculations for the local thermal flux distributions was developed. The microscopic thermal-group cross sections are modified by taking flux-weighted averages of these constants over the cell area. The modified microscopic cross sections are then multiplied by the same volume-averaged atomic number densities used in calculating the fast group constants. The result is a set of homogenized macroscopic thermal group cross sections which are used in the two-dimensional reactor calculations.

The validity of this flux-weighting technique was evaluated for a seed-blanket configuration made sufficiently small to permit trial by a two-dimensional calculation which explicitly described the geometric heterogeneity. For the case examined, the values calculated for k_{eff} and blanket power fraction were not affected by replacement of detailed geometric structure with a combination of flux-weighted constants and a homogenized representation of the core.

Consequently, this technique has been used to provide the thermal-group constants used in the refined two-dimensional calculations for the peak-to-average power generated in each reactor region, and in one-dimensional calculations for the total power fraction generated in each of these regions.

(b) Representation of Seed Clusters Having Control Rods Inserted. The presence of control rods in particular seed clusters is described by the addition of a uniform thermal poison to the constants which describe the rodded clusters. This technique is valid only to the extent that the true neutron absorption in control rod material can be described as being a thermal absorption effect alone. In the case of the hafnium material used in the control rods of the PWR, it is known that a substantial fraction of the control effect is provided by the

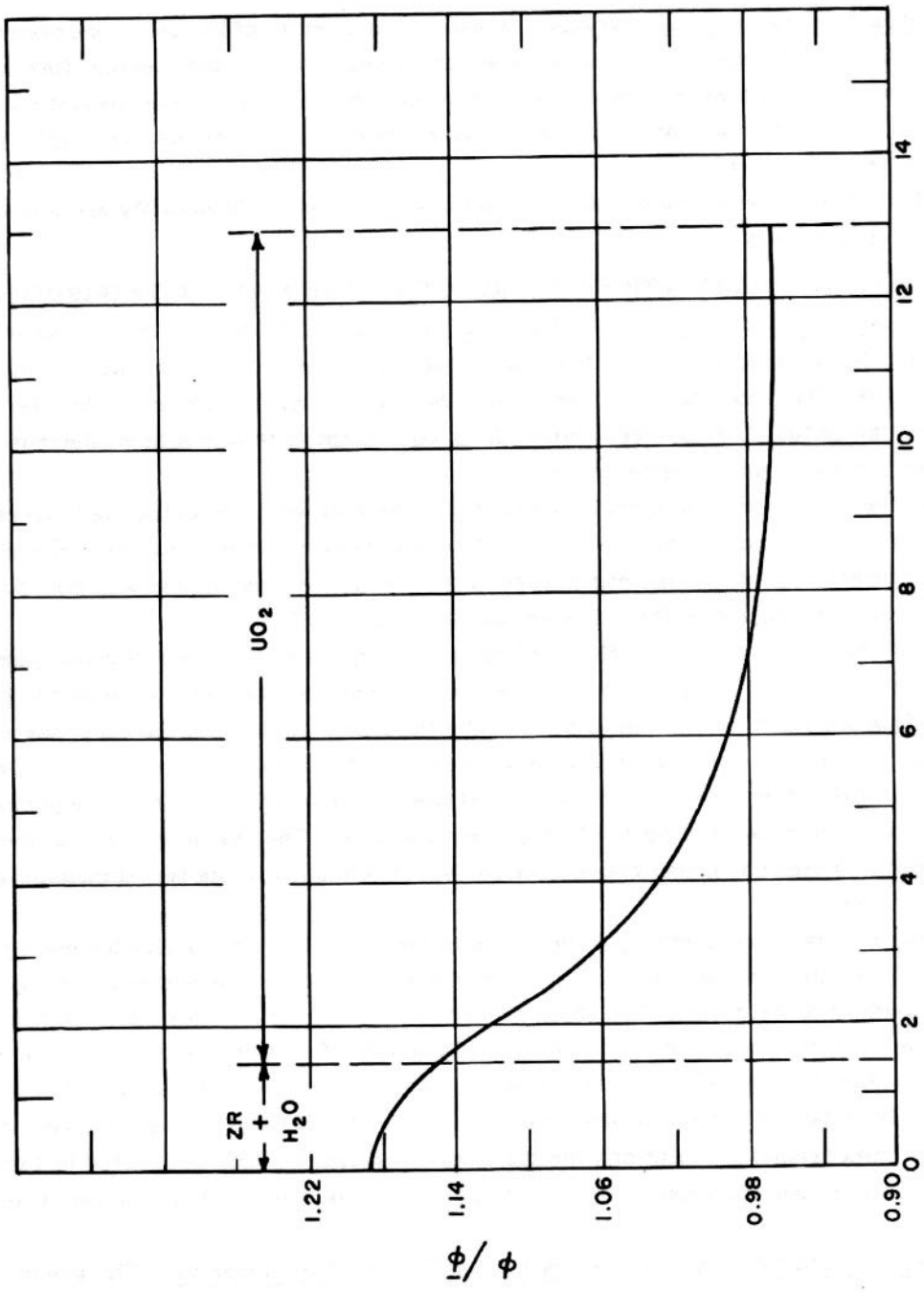


Fig. 4.14 - Axial Thermal Flux Traverse Through Blanket Rod and Tube Sheet. Blanket Average Flux Normalized to 1.0.

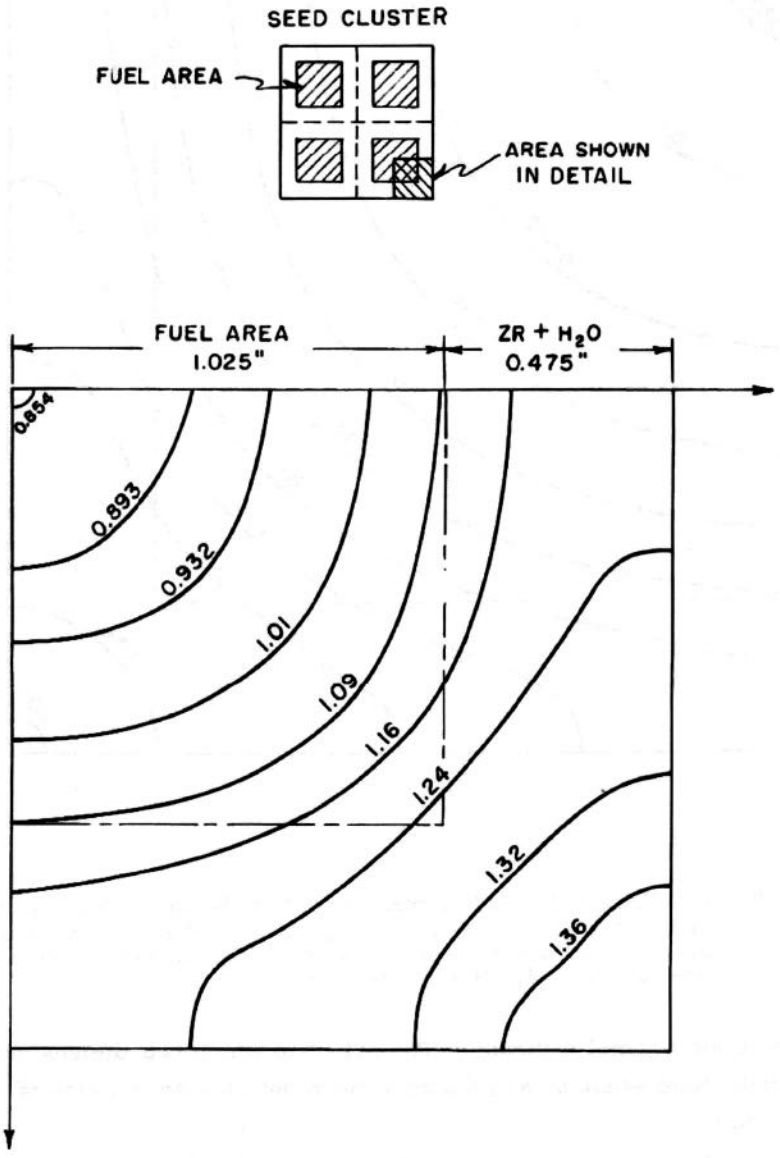


Fig. 4.15 - Two-Dimensional Thermal Flux Distribution in Seed Calculated for Reference Design at Operating Temperature. Distribution Normalized to Average Flux in Fuel Region.

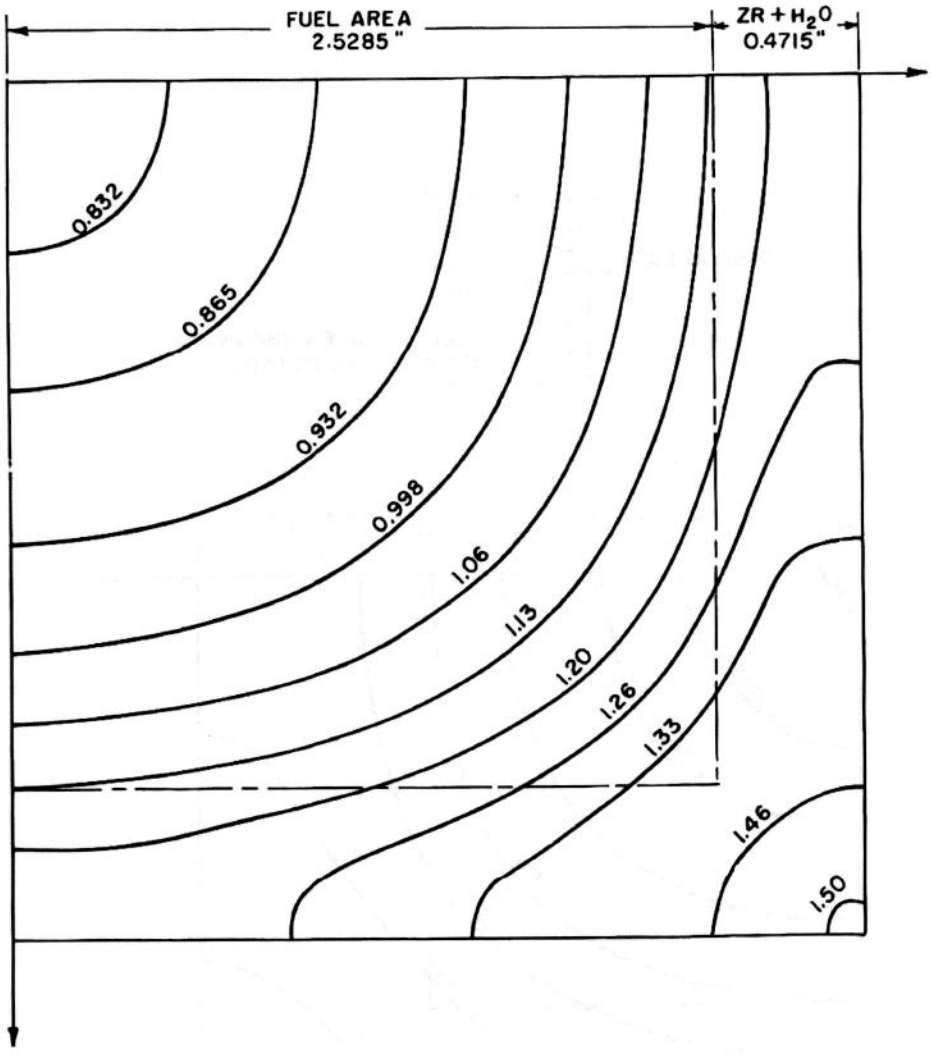


Fig. 4.16 - Two-Dimensional Thermal Flux Distribution in Blanket Bundle. Calculated for Reference Design at Operating Temperature. Distribution Normalized to Average Flux in Fuel Region. One-Quarter of a Blanket Bundle Shown.

resonance capture of epithermal neutrons. The extent to which two-dimensional calculations for the power distributions would be altered by a more detailed description of the control rods is not presently known.

(c) Peak-to-Average Power in the Seed and Blanket Regions. As indicated in Sec. 4.5, the results of the two-dimensional reactor calculations are used principally to provide a reasonable estimate of the radial power peaks to be expected in that section of the reactor where the axial power shape is a maximum. Thus, the information presently used to predict the flux distributions and the thermal performance of the reactor has been based in part on a series of such two-dimensional calculations in which quadrant symmetry and various control rod configurations were assumed.

(d) Core Power Fraction in Various Regions. One-dimensional radial flux distributions obtained by adjusting the reactor to criticality with a uniform thermal poison added to the seed

have been used to describe the fraction of power generated in each core subregion. This follows the interpretation given to the results of one-dimensional reactor calculations as presented in Sec. 4.5.

(e) Reactivity Lifetime Calculations. Calculations for the k_{eff} of the core in its clean condition, either cold or hot, are obtained by a two-dimensional calculation of the type described above, with no control rods inserted into the core. The results of such a calculation is an eigenvalue which is interpreted to be the value for k_{eff} applicable to the core when in the condition described by the calculation. The use of flux-weighted thermal constants and three fast neutron energy groups appears to result in values for k_{eff} which are consistently 0.01 to 0.02 higher than the value obtained from critical experiments. These critical experiments were all performed near room temperature using slab seed and blanket critical assemblies (see Chap. 9, Sec. 9.32). A bias of this same order of magnitude has been assumed to apply to the calculations for the hot reactor because no experimental information is available which would provide the basis for a better estimate.

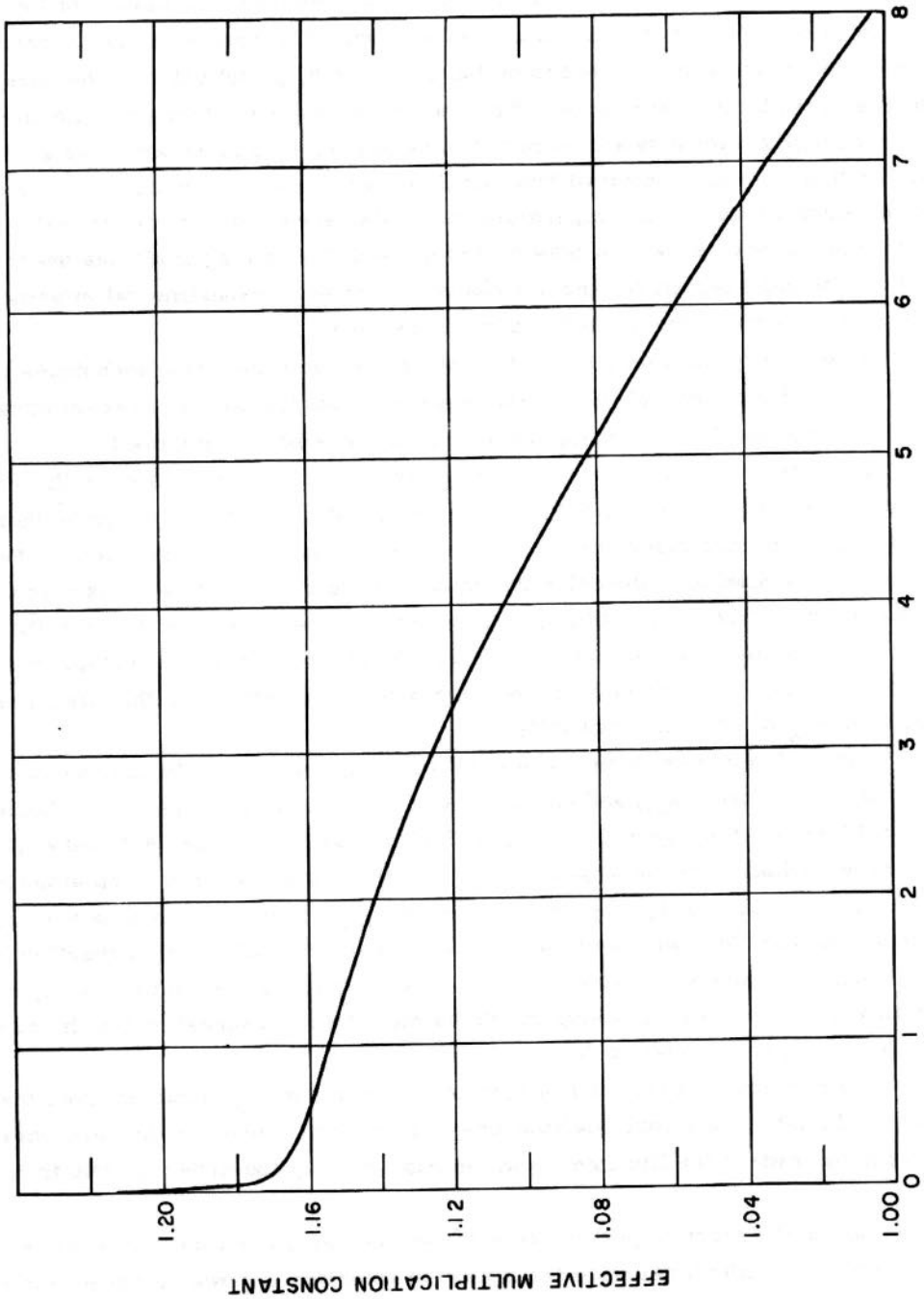
Calculations of reactivity lifetime could not be performed with the same techniques used for the clean reactor. This analytical limitation resulted from the lack of a two-dimensional depletion code at the time the detailed design of the reactor was being developed.

The most advanced reactivity lifetime code available for the detailed design of the PWR was CANDLE (see Appendix C). The CANDLE code permits the use of flux-weighted thermal group constants for a clean core calculation. However, since no obvious extension of the flux-weighting technique was available to describe the change in local flux distributions with spatially dependent fuel depletion, it was not possible to utilize the flux-weighting technique in the reactivity lifetime calculations. Consequently, the reactivity lifetime of the reactor is calculated for a one-dimensional representation of the core with no allowance for the effects on power distribution or k_{eff} of core heterogeneity.

Figures 4.17 and 4.18 summarize the results of a four-group reactivity lifetime calculation in terms of the variation of k_{eff} and blanket power fraction with lifetime. (It should be noted that Figs. 4.17 and 4.18 apply to the detailed PWR design while Figs. 4.9 and 4.10 apply to early survey studies.) An improved estimate of the actual core life is obtained by correcting the calculated values of k_{eff} by the ratio of the k_{eff}'s from hot, clean one-dimensional calculations with and without flux-weighted thermal constants. An additional correction factor is applied in estimating the core reactivity lifetime by recognizing that even the four-group calculation with flux-weighted thermal group constants consistently overestimates the experimental value for k_{eff} by approximately 0.02.

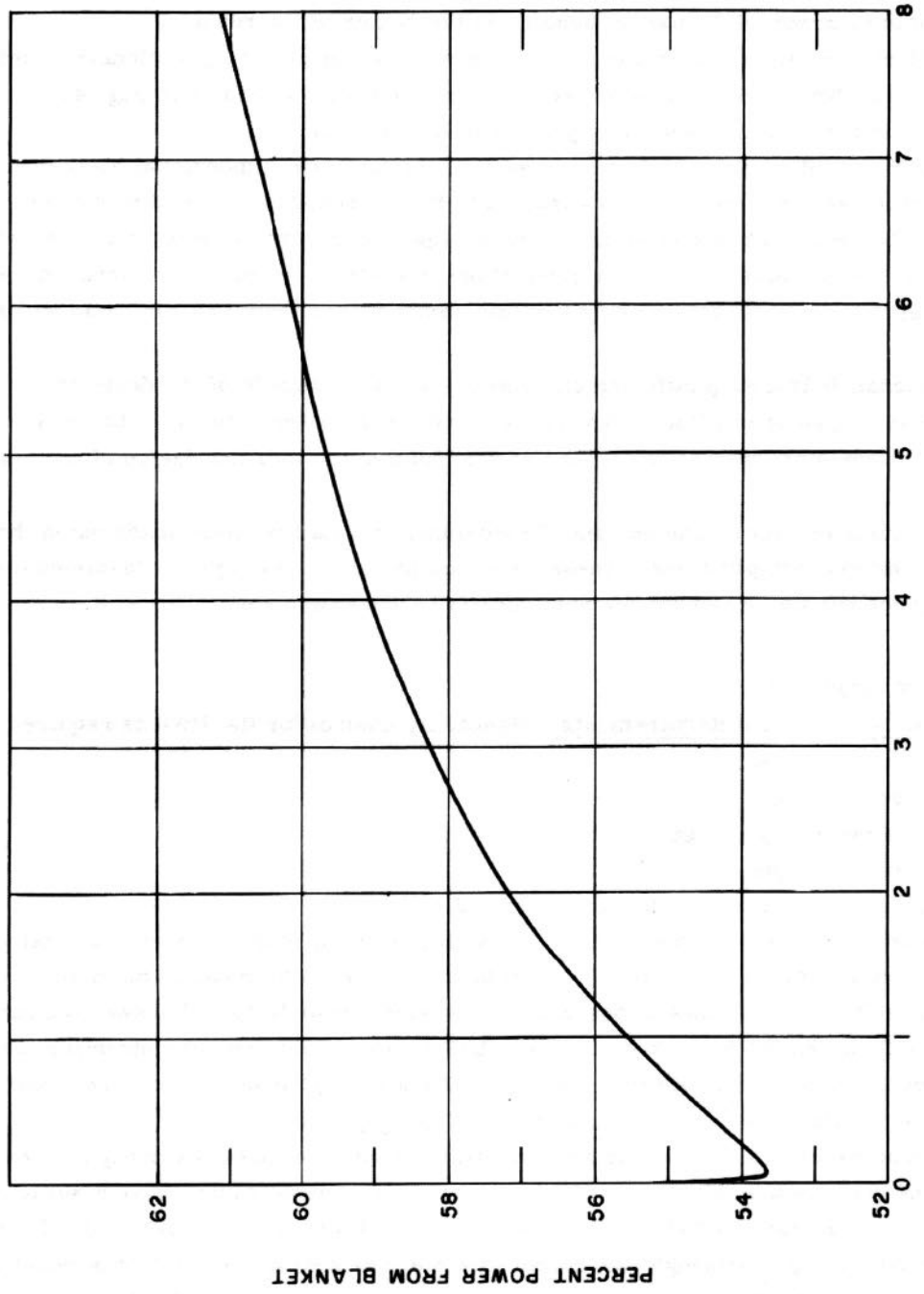
The result of applying these correction factors to the values of k_{eff} obtained from the reactivity lifetime calculation is a core lifetime which is presently taken as the reference value. The corrections reduce the lifetime shown on Fig. 4.17 by the order of 2000 to 3000 hours.

In order to establish the thermal performance of the core as a function of reactivity lifetime, data must be obtained which describe the timewise variation of power fraction and gross radial peak-to-average power ratio in each subregion of the reactor. The absence, at the time of the detailed design study of the PWR, of a two-dimensional reactor code to describe explicitly timewise variations in isotopic composition necessitated the use of an approximate method for obtaining the peak-to-average ratios.



PWR reference design. Seed loading 75 kg. Four-group, one-dimensional cylindrical calculation with CANDLE. Not corrected for flux-weighted constants or criticality bias.

Fig. 4.17 - Variation of Core Reactivity with Reactor Lifetime.



PWR reference design. Seed loading 75 kg. Four-group one-dimensional cylindrical calculation with CANDLE.

Fig. 4.18 - Variation of Blanket Power Fraction with Reactor Lifetime.

The method used was to divide the blanket into four regions. These regions were as follows:

1. The outer row of 40 blanket bundles immediately adjacent to the seed.
2. The inner row of 24 blanket bundles immediately adjacent to the seed.
3. The outermost row of 28 blanket bundles.
4. The remaining group of 21 blanket bundles at the center of the reactor.

The first three blanket regions were made to correspond to an annular ring of blanket material in the cylindrical one-dimensional model of the reactor by equating areas as in Fig. 4.5. The central, cylindrical region was similarly made to correspond.

The isotopic composition during lifetime in each of the four one-dimensional blanket regions was averaged and this composition assigned to the corresponding two-dimensional blanket region. The seed loading was similarly made equal in the two-dimensional case to the one-dimensional depleted value. Thus at a given stage of lifetime, a two-dimensional model of the depleted reactor could be set up using isotopic compositions from one-dimensional lifetime studies.

Two-dimensional, four-group diffusion calculations were then made of the depleted reactors at various stages of lifetime with various control rod configurations. The resulting two-dimensional power distributions were used as the basis of the thermal design of the reactor during lifetime.

As a partial check on this technique, one-dimensional calculations were made using the region averaged isotopic compositions. These were compared to the original one-dimensional results. The agreement was excellent for both reactivity and power generation in the subregions.

4.9 REACTIVITY CONTROL

(a) Reactivity Control Requirements. Reactivity control for the PWR is required for the following specific reasons:

1. Cold shutdown margin
2. Cold to hot reactivity change
3. Equilibrium xenon poison
4. Fuel depletion and stable fission product build-up

The cold shutdown margin provided by the 32 hafnium control rods in the seed clusters of the reactor has been subjected to a special requirement: The reactor must be maintained in a subcritical state in the extreme case of one control rod stuck in its fully withdrawn position. Measurements made in the PWR mock-up indicate that the reference control rods will provide this one stuck rod criterion. Furthermore, with all 32 rods fully inserted, the cold, clean reactor appears to be shut down by approximately 0.07 in k_{eff}.

The cold, clean reactor must provide a substantial amount of excess reactivity in order that the k_{eff} of the core, which decreases with increasing core temperature, have a sufficiently high value in the hot condition so that the required reactivity lifetime can be attained. This cold to hot reactivity change, although associated with the temperature coefficient of reactivity, is not related to that quantity in any simple manner. Since no experimental facility is now available which would allow direct measurement of the temperature defect, its magnitude has been estimated by calculating the total k_{eff} of the clean reactor in the cold and in the hot condition. The value for the temperature defect for the PWR is presently estimated to be between 0.035 and 0.040 in k_{eff}.

The effect on k_{eff} of fuel depletion, plutonium build-up, and the build-up of the fission product poisons has been determined by analysis of the one-dimensional reactivity lifetime calculations. The variation of k_{eff} with reactivity lifetime as described in Fig. 4.17 indicates that the change in k_{eff} caused by the build-up of equilibrium xenon is approximately 0.04. The rate of change of k_{eff} with time caused by fuel depletion and stable fission product poison build-up is approximately 0.02 per 1000 hours of equivalent full power operation.

An estimate of the PWR reactivity balance, in terms of Δk_{eff}, is as follows:

1.	Cold shutdown margin	0.07
2.	Cold to hot reactivity change	0.04
3.	Equilibrium xenon poison	0.04
4.	Lifetime depletion	0.12
	Total Δk_{eff}	0.27

(b) Control Stability - The Temperature Coefficient of Reactivity. Extensive studies have shown that a substantial negative temperature coefficient of reactivity is an extremely important factor in providing over-all plant stability and ensuring safety of the reactor in the event of certain types of accidents. Simulator studies of such accident conditions as loss of coolant flow or the inadvertent addition of reactivity, particularly during reactor startup, have shown the importance of the negative temperature coefficient in protection of the reactor.

For these reasons, the PWR has been designed to have a substantial negative temperature coefficient at operating temperature. While no completely reliable method for calculating the temperature coefficient has been developed, it has been estimated by calculating the k_{eff} of the reactor at two distinct but slightly different temperatures. An estimate for this coefficient in the hot reactor has been obtained. It is expected that the temperature coefficient of reactivity for the PWR will be between -2×10^{-4} and -4×10^{-4} in reactivity per degree Fahrenheit at the operating temperature of 525°F

(c) Control Rod Programming. Because the control rods represent localized regions of high neutron absorption properties, the spatial distribution of the neutron flux will be influenced by the control rod arrangement. Thus a relationship exists between the control rod configuration at any time and the thermal capability of the reactor. A method for determining this effect of the control rods on the flux distribution and consequently on the thermal performance of the core has been discussed in Sec. 4.8.

As a result of such studies, an optimized control rod programming sequence was established which provided for the sequential withdrawal of several groups of rods, one group at a time, in order to make up the continuing loss in reactivity which accompanies power operation of the reactor.

The order of withdrawal of control rod groups can be described as follows: On Fig. 4.1 designate the seed clusters on one side of the seed by numbering them 1 through 7 starting with the end cluster. Designate the inset cluster as No. 8. The clusters in the other sides are similarly numbered up to eight. The order of withdrawal is as follows:

Group 1 - 16 rods - Nos. 1, 3, 5 and 7
Group 2 - 8 rods - Nos. 2 and 6
Group 3 - 4 rods - No. 4
Group 4 - 4 rods - No. 8

It is expected that hot, clean criticality will be reached with Group 1 almost full withdrawn. In power operation, with xenon present, Group 2 would be partially withdrawn. The remaining groups would be withdrawn as required for depletion and poisoning compensation.

An alternate mode of control rod operation which was considered for the PWR design involved the simultaneous motion of all 32 rods. This type of control, called banked rod control, maintains all rods at the same axial position. For the PWR it was found to provide less favorable values for thermal performance than the sequential type of programming which was chosen.

(d) Flux Tilting. Because of the relatively large dimensions of the PWR, compared to the average distance a neutron can travel during its lifetime, it is possible that different regions of the reactor could, under certain conditions, behave like small, independent reactors, each with its own potential for a sustained, steady-state chain reaction. This loose coupling of the different sections of the core makes possible a spatial unbalance of the neutron distribution. In particular, an azimuthal variation in flux level around the seed is possible. Any tendency of the reactor power level to tilt must be avoided since this would be prejudicial to maintaining rated power output within design temperature limits.

A flux tilt might arise because of structural or compositional differences among the fuel assemblies, because of a non-symmetric distribution of coolant temperatures in the core, because of an asymmetric arrangement of the control rods, or because of different levels of xenon poisoning around the reactor. From the results of experimental and analytical studies of this problem, it was concluded that non-uniformities among the manufactured fuel assemblies were not great enough to produce a measurable tilt. The possibility of tilting caused by other factors, however, does remain and must be controlled.

The principal problem in controlling flux tilting is detecting the presence of the tilt. In the PWR this problem has been met by locating suitable neutron detectors at symmetric positions around the periphery of the pressure vessel and by including thermally instrumented fuel assemblies in the core. It is expected that relatively small control rod motions will remove any flux tilts which develop.

4.10 LONG TERM REACTIVITY CHANGES

Long-term reactivity changes in material high in U^{238} content are not well understood for several reasons. First, there is a lack of detailed information on the absorption and fission cross sections of various isotopes, particularly in the thermal and epithermal region. Second, the calculation of the neutron distribution in energy at various positions in the reactor is very difficult. The problem is outlined here to the extent that it is presently understood.

The production of Pu^{239} in the blanket compared to the rate of destruction of U^{235} varies with location in the blanket because the ratio of fast to slow neutron flux varies spatially. This variation arises from the over-all flux distributions between seed and blanket and because of local heterogeneities.

(a) The Conversion Ratio. The conversion ratio (C. R.) is defined here to be the ratio of fissionable material production rate at a certain point to the fissionable material destruction rate at that point. If the Pu^{240} concentration is low, this ratio for the PWR blanket may be written

$$\text{C. R.} = \frac{\text{Absorption of neutrons in } U^{238}}{\text{Absorption of neutrons in } U^{235} + \text{Absorption of neutrons in } Pu^{239}} \qquad \text{Eq. (4. 2)}$$

It is assumed here that essentially all absorption in U^{235} and Pu^{239} occurs at thermal energies. Production of Pu^{239} arises from neutron capture in U^{238}, both at thermal energies and in the epithermal U^{238} resonances.

The conversion ratio for the clean reactor may then be more simply expressed as

$$\text{C. R.} = \frac{p^{28}(\Sigma_a^{28}/\Sigma_a^T) + (1 - p^{28})}{p^{28}(\Sigma_a^{25}/\Sigma_a^T)} \qquad \text{Eq. (4. 3)}$$

In this expression the symbols have the following significance: The resonance escape probability, p^{28}, describes the fraction of neutrons entering the blanket region above the U^{238} resonances which are slowed to thermal before capture, and Σ_a^{25}, Σ_a^{28}, and Σ_a^T are the macroscopic thermal absorption cross sections in U^{235}, U^{238}, and all materials, respectively.

The argument used in deriving Eq. (4. 3) is the following. Leakage of neutrons in the resonance and thermal region is neglected. For every fast neutron entering the blanket, a fraction $(1 - p^{28})$ is captured in U^{238} resonances, and p^{28} neutrons reach thermal, where a fraction Σ_a^{25}/Σ_a^T is absorbed in U^{235}, and a fraction Σ_a^{28}/Σ_a^T is absorbed in U^{238}. In the clean core no Pu^{239} is present, and Eq. (4. 3) follows directly from the definition of conversion ratio.

For a simple analysis, Eq. (4. 3) offers the advantage that conditions of criticality and neutron leakage need not be considered. Insertion of the hot clean values for the PWR blanket gives a value for the conversion ratio of approximately 1. 13. Almost half of this conversion is due to capture in U^{238} at thermal energies.

The conversion ratio calculated by Eq. (4. 3) is essentially the asymptotic value for the blanket material. Where there is a strong neutron leakage, either inward or outward, the conversion ratio will differ from that predicted by Eq. (4. 3). In particular, near the seed-blanket interface and in the blanket assemblies close to the 1/2-inch-thick water channels the value of conversion ratio is substantially lower than the average value calculated for a homogenized blanket region. This is graphically illustrated in Fig. 4. 19 where the conversion ratio is plotted as a function of position in the blanket of a slab core of seed and blanket material. The reactor is composed of one row of seed clusters with two rows of blanket bundles on each side. The constants are representative of the PWR reactor in its hot, clean condition.

Calculations of the blanket conversion ratio based on data obtained from reactivity lifetime calculations indicate that the average value of this quantity becomes smaller as the blanket is irradiated, and falls from its initial value of 1. 13 to about 1. 00 after 3000 equivalent full power hours of core operation.

The homogenized average conversion ratio is computed implicitly in the reactivity lifetime calculation. To the extent that the isotopic cross sections for the important materials are known, a first estimate of the reactivity effects to be expected from irradiation of natural uranium has been included in the results summarized by Figs. 4. 17 and 4. 18.

(b) Effects of Plutonium Build-up on Power Peaking Factors. In the hot, clean reactor the local distribution of power in a blanket assembly is peaked at the corners adjacent to the water channels. The information in Fig. 4. 19 implies that the effect of plutonium build-up might be a reduction in magnitude of that flux peak, since the conversion ratio is highest at the center of the assembly where the power is at a minimum, while the depletion of U^{235} will be greatest at the initial location of the power peaks. However, since the local production rate

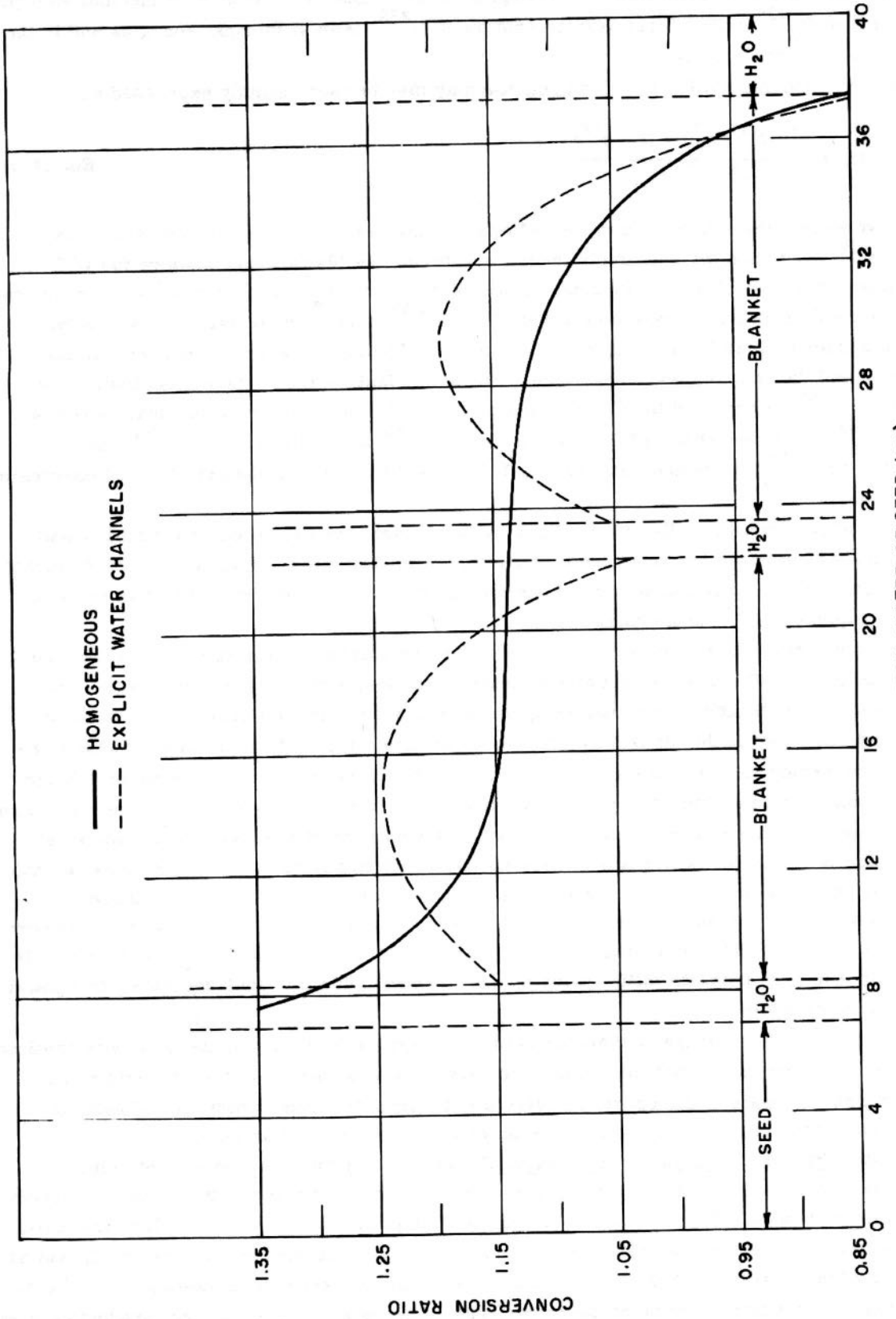


Fig. 4.19 - Conversion Ratio as a Function of Position for a Slab Geometry Seed and Blanket Reactor. (Hot, clean PWR constants).

of Pu^{239} is given as a product of local conversion ratio times the fissionable material burnup rate, the peak-to-average power in the assembly may actually increase with time. No detailed analytical investigation of the time-varying peaking factor has been made for the PWR.

4.11 EVALUATION OF EXPERIMENTAL RESULTS

The PWR reactor design was supported by a complete mock-up type critical facility, the PWR Flexible Assembly. This facility and the experiments are described in Chap. 9, Secs. 29 through 31. The results of the program of studies with slab reactors and a full-scale mock-up reactor are summarized in Sec. 9.32. Rather than repeat these results, some comments will be made by way of summary and evaluation.

(a) Criticality Studies. Criticality studies were made of various slab reactor arrays with varying amounts of blanket. In addition, criticality studies were made of the full-scale mock-up. These criticality studies were made with no control rods in the core; criticality was achieved by dropping the moderator height to an appropriate value. The effective axial buckling was determined by an axial flux traverse. In the case of the slabs, the transverse buckling along the length of the slab was also determined by a flux plot.

With these data the design type calculation was tested to find the eigenvalue predicted for the experiment. The material inventory of the particular reactor was used to determine the reactor parameters. The MUFT calculation was performed to obtain the fast-group constants. The thermal-group constants were determined by the flux weighting technique described in Sec. 4.8a above, together with the use of the SOFOCATE code to determine the appropriate spectral averages.

The few-group constants determined in this way were then used in either a one-dimensional or two-dimensional spatial integration, depending upon the nature of the reactor, i.e., slab or full mock-up. The calculations were performed in both two-group and four-group models.

The four-group calculation for the k_{eff} of a critical reactor composed of PWR mock-up materials resulted in a value which ranged from 1.01 to 1.02 rather than the critical value of 1.00 (see Table 9.28). The value of k_{eff} for the corresponding two-group case was typically 1.04 rather than 1.00. This bias did not appear to be associated with any random error in the experiment but was probably a systematic error associated with the theoretical model.

As a consequence of these results it is assumed in PWR design considerations that a four-group calculation of the reactor properties will overestimate the k_{eff} by about 2 per cent.

(b) Excess Reactivity Measurements. The experimental technique for measurement of the excess reactivity of a reactor by the differential water height technique is described in Chap. 8, Sec. 19. The results obtained are given in Chap. 9, Table 9.28.

The interpretation of the excess reactivity experiment involves the use of the inhour equation. Thus, total yield of delayed neutrons and partial yields at various periods are involved in this method. Because of the uncertainty in these yields, the interpretation is ambiguous (see Sec. 8.23).

The excess reactivity of the PWR mock-up reactor as predicted by the design calculation was 17.3 per cent. The values of the excess reactivity measured on the full-scale mock-up was either 16.3 per cent or 17.5 per cent, respectively, depending upon whether Keepin's data or Hughes' data on delayed neutron fractions were used.

(c) Shutdown Measurements. The technique for determination of the reactor shutdown was primarily the negative buckling technique described in Chap. 8, Sec. 21. While the rod

drop technique and the rod drop calibration method were used to a certain extent, principal reliance was placed on the data obtained by the negative buckling technique. The results obtained by the other methods agreed sufficiently well with the negative buckling results, so that it was clear that no serious disagreement existed. The negative buckling technique was sufficiently reliable that on certain experiments in which small changes in the reactor material buckling were made which affected the excess reactivity, a corresponding change in shutdown reactivity was observed.

The shutdown provided by the control rods used in the mock-up was about 8 per cent in reactivity (or about 7 per cent in k_{eff}). The control rods used were made of 30 w/o cadmium, 70 w/o silver, and had the same span as the hafnium control rods used in the design core. Their thickness was 0.250 inch rather than the 0.220 inch thickness of the hafnium rods. In a separate series of experiments, the hafnium rods were compared with cadmium-silver rods, considered either as individually programmed rods or in a banked configuration. It was found that the hafnium rods were worth no less than the cadmium-silver rods used in the mock-up, and possibly were worth more by as much as 3 per cent of their total worth.

(d) Control Rod Worth Measurements. It should be pointed out that no attempt was made to specify the PWR control rod span by analytic techniques. Instead, an estimate was made early in the conceptual design stage of the control rod span required. This size was then modified somewhat to be as large a span as could be accommodated with mechanical convenience on the design. The detailed mechanical design and the full-scale mock-up construction proceeded on the basis of this span size (3-3/8 inches).

For nuclear design purposes the control rod worth was taken to be a thermal poison which was distributed uniformly over the seed cluster containing the control rod. The value chosen for this poison represented an appropriate shutdown condition for the cold reactor. The value of this macroscopic poison absorption cross section was 0.05 cm^{-1} in the cold condition.

In the mock-up studies of the reactor shutdown, the negative buckling technique yielded values for the homogenized thermal poison worths of the control rod bank. The values obtained varied between 0.04 and 0.05 cm^{-1}.

A direct experimental determination of the thermal poison equivalent of the control rods was also made. Boron loaded plastic tapes, distributed throughout the seed region in as uniform a fashion as possible, were substituted for the control rods. The quantity of boron required to provide the same control as that provided by a single isolated control rod was determined, as well as the quantity required to represent the same rod in a banked rod configuration. These experiments were difficult to interpret because of the heterogeneous nature of the experiment and the difficulty of distributing the poison uniformly. The values of Σ_p, the homogenized macroscopic boron thermal absorption cross section obtained in these experiments, were approximately 0.040 cm^{-1} for the single isolated control rod and 0.044 cm^{-1} for the control rod in a bank. The estimated uncertainty in these results is about 15 per cent.

The conclusion to be drawn from these results is simply that the chosen control rod span was approximately correct, since it gave the desired design properties.

(e) Conversion Ratio Measurements. Measurements of the conversion ratio in the blanket of the cold mock-up reactor were performed, using the technique described in Chap. 11, Sec. 14. The spatial variation and magnitude of the measured conversion ratio agreed quite well with a one-dimensional four-group calculation which described explicitly the water channel

between the seed-blanket interface[4]. A one-dimensional homogeneous calculation which homogenized the water channel into the seed and blanket gave an average value of the conversion ratio that agreed fairly well with the experimental result. A two-dimensional four-group calculation gave results approximately 5 per cent higher than the experimental values of the conversion ratio. The reasons for this disagreement between the one- and two-dimensional calculations are not known.

(f) Measurements of Gross Flux Distribution. Experiments on over-all flux distribution are, at present, preliminary in nature and no definite conclusions can be drawn. Some experiments have been performed on the mock-up in an attempt to compare the over-all gross flux distribution between the seed and the various blanket orificing regions. These experiments indicated that the ratio of power between seed and blanket was predicted by a homogenized calculation within an uncertainty of approximately 20 per cent. This large uncertainty was associated with the difficulty of comparing the measured heterogeneous flux distributions with the homogeneous calculation. The distribution of power between the various blanket regions could be determined with somewhat better precision than could the seed-blanket power sharing since, to a first approximation, the heterogeneous blanket flux shape is a superposition of a homogeneous distribution and a blanket cell distribution. The experimental results indicated that the simplified two-group technique of calculation used in the earlier nuclear design work did not adequately describe the way in which the power fell off in regions remote from the seed. These calculations predicted a more rapid attenuation of the power level away from the seed than the experiment indicated. The four-group calculations agreed more closely with the experimental data in this respect. Thus, as far as can be determined at present, the use of four-group calculations appeared necessary to provide a proper description of the general behavior of the blanket power distributions in regions remote from the seed.

4.12 SUMMARY OF DESIGN

The PWR reactor has a fuel loading of 75 kilograms of U^{235} in the seed region. With this fuel loading, the reactivity lifetime of the reactor is estimated to be approximately 6,000 equivalent full power hours. The calculations indicate that somewhat over 50 per cent of the total power will be developed in the blanket. Based upon the calculations and the analyses of the experiments for the cold mock-up reactor, the excess reactivity of the cold, clean core is estimated to be approximately 17 per cent. The control rods are estimated to shut the reactor down by 7 per cent in k_{eff}. The calculations indicate the value for the temperature defect of the reactor is 3.6 per cent in k_{eff}, and that the reactivity loss due to the build-up of equilibrium xenon is 4 per cent in k_{eff}. The loss in k_{eff} due to the effect of fuel depletion and stable fission product poison build-up can be determined by differences to average two per cent per 1000 hours of full power operation.

The temperature coefficient of reactivity for the PWR at operating conditions has been calculated by the method mentioned in Sec. 4.9b. The temperature coefficient obtained was $-3 \times 10^{-4}/°F$.

It should be noted that these results represent the best estimates of the characteristics of the first PWR core approximately six months in advance of operation.

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Pages 162 - 376 have been redacted in their entirety.

Part III

TECHNIQUES OF REACTOR PHYSICS EXPERIMENTATION

This portion of the Handbook describes the methods used in reactor physics experimentation. The techniques of reactor physics experimentation are in essence the techniques of exponential and critical experiments. At both Bettis and KAPL the critical experiment has been used exclusively. Some important work with slightly enriched uranium reactors moderated with ordinary water has been done at Brookhaven, using an exponential experiment.

In Chap. 8 a general description of critical experiment design for water moderated reactors is presented. The requirements of the experiment from both a functional and safety point of view are given. Following this discussion the techniques of experimentation are described in a general fashion. In Chap. 9 a description of certain critical facilities is given as well as the manner in which they have been used. The specific features of the DCTF (Danger Coefficient Test Facility), the STR-CA (Submarine Thermal Reactor - Critical Assembly), the TRX (Two Region Critical Experiment), the SFR critical experiment, the SAR critical experiment, and the PWR critical experiment are described. Typical experimental data are included in the presentation to clarify the discussion.

Chapter 10 is a general discussion of the techniques of flux mapping in critical assemblies. The problem of flux mapping of reactors is of such importance in critical experiment work, and has required such extensive development, that an entire chapter has been devoted to it.

Chapter 11 presents the techniques used in measurement of resonance escape probability, fast fission factor, and thermal utilization. These parameters are of particular significance in slightly enriched reactors. Considerable effort has been necessary to bring these measurement techniques to their present state of development.

Chapter 8

CRITICAL EXPERIMENT DESIGN AND TECHNIQUES

GENERAL DESIGN FEATURES OF WATER MODERATED CRITICAL ASSEMBLIES

by F. J. Jankowski

8.1 INTRODUCTION

Critical assemblies have proved to be important tools for the design of power reactors. The most important requirements of a critical assembly for reactor design purposes are safety, flexibility, and ease of operation. While cost is an important factor, it must be considered in the light of various choices open to the nuclear designer.

In critical experimentation for the design of pressurized water power reactors, the use of water at atmospheric pressure as the critical assembly moderator is extremely attractive. It meets the requirements of safety, flexibility, and ease of operation. Because of the importance of water, this chapter will be devoted exclusively to a discussion of techniques of water moderated critical experiments.

The nuclear properties of water at elevated temperatures are significantly different from those of water at room temperature. Water moderated critical assemblies operated at atmospheric pressure are, however, limited in the temperatures to which the water may be heated. To overcome this difficulty, ways have been sought to simulate the elevated temperatures at which pressurized water reactors are operated. These attempts include the use of other liquid hydrogenous moderators such as furfuryl alcohol and plastics. The differences in critical experiment design for the use of other hydrogenous liquids are slight. Some of the features of plastic-moderated critical assemblies are covered in Chap. 9.

8.2 CRITICAL EXPERIMENT CORE DESIGN

Critical experiments may be considered to fall into three classes. The first type is essentially a proof test of a manufactured design; in this type of experiment a great deal of attention is given to duplication of the design details of the power reactor. Such experiments require the least degree of flexibility. The next class is the mock-up type which must be a good approximation to the nuclear design of the reactor, but which sacrifices detail for flexibility. Finally, there is the clean critical type of experiment in which geometrical

simplicity and homogeneity are important factors. The functions of these classes of experiments are quite different. The clean critical experiment is used to provide basic information about the reactor. The mock-up permits full scale design studies while the design is undergoing changes. The proof test reactor is a final test of the fully developed design. In many cases a facility must be designed to permit either mock-up or clean critical work. The proof test reactor may be bypassed through use of special subassemblies in the mock-up.

The factors for which design flexibility must be provided are:

1. core size
2. core geometry
3. fuel loading
4. metal-to-uranium loading
5. metal-to-water ratio
6. cell size (for natural and slightly enriched reactors)
7. control rod size and location

Depending upon the particular reactor, some of these quantities may be relatively fixed while others are relatively uncertain. The flexibility to be provided for each factor must be considered in making the mechanical design.

Safety requires that the design provide for adequate strength of the core and its components. Rigidity is required so that unaccountable variations in reactivity will not occur to prejudice an experiment. The possibility of air entrapment in the core should be considered. This can be a troublesome source of reactivity fluctuations. When studies are to be made at varying temperatures, the circulation of water through the core must be considered in fixing the mechanical design.

Protection of the core elements from corrosion is, in some cases, a problem in water moderated reactors. Generally speaking, the problem is that of avoiding contamination of the water by radioactivity. While corrosion by cold water is not as serious as that by hot water, the general problem must be considered in choosing a fuel element design. Uranium and some uranium alloys are particularly subject to corrosion. These materials can be protected by cladding with plastic such as polyethylene. The cladding is a hydrogenous substance similar to water, and therefore does not affect the core greatly. Foreign metallic platings not present in the power reactor such as nickel, lead, tin, or zinc may be used if the total amount is kept low. Metals have also been protected by jacketing. Aluminum jacketing of uranium, and aluminum and zirconium cladding of their respective alloys have proved feasible. Another important consideration in cladding, plating, or jacketing is reducing the contamination of water by recoil fission products.

Safety considerations require that two ranges of shutdown must be provided in a critical experiment. The first, generally obtained by control rods, enables the reactor to be shut down with some adequate margin of reactivity, for example 5 per cent. The second range, the so-called safety shutdown, permits major modifications to be made in the core. In water moderated reactors, safety shutdown is achieved by draining the liquid. In designing the core, however, care must be taken to limit the amount of plastic because this can limit the available range of safety shutdown. In plastic moderated cores, safety shutdown is achieved by separating the core into two or more sections.

To obtain the desired degree of flexibility in the reactor core, a modular type construction is generally used. The fuel and structural material is assembled into units of 1 by 1 inch,

2 by 2 inches, or some similar dimension. The length of the unit is generally the height of the core, but this can also be built up of fractional lengths. These units, or subassemblies, are made up in an area outside the reactor assembly room (the staging area) and are loaded into the core as needed. By having an excess of subassemblies available, it is possible to start modification on a core while critical experiments are continuing.

The size of the module to be used is primarily determined by the features of the reactor to be built. The size of the reactor, the nature of the variations which must be made, and the dependence of the nuclear properties upon the fine structure of the core are important considerations. Small modules have the advantages of greater flexibility in core geometry, greater freedom in location of control rods, and lighter and more easily handled subassemblies. There are, however, more pieces to be handled; and core rigidity may be a problem. The larger modules may sometimes permit a better approximation to the reactor design than the small modules.

8.3 CONTROL RODS

The design of the control rod depends on the type of critical assembly being built. A clean critical, where it is desired to have no perturbations from rods, rod guides, or rod channels, requires a special type of control rod. For this purpose, there are a number of possible designs. One is a thin blade or cross-shaped rod that fits between the fuel assemblies or in the moderator channels. Another, is to have the control rod displace fuel from the core.

In the mock-up critical assembly the control rod is similar in shape and size to that planned for the final reactor. The reactivity worth of a control rod depends on the rod material, size, and location as well as on the fuel loading of the core and on the neutron spectrum in the core. At the present time, it is difficult to predict the worth of a control rod with accuracy. For this reason, substitute rod material may be used in a mock-up assembly for preliminary studies, but the actual rod materials should be used for final assemblies.

A number of materials are available for use as control rods. Perhaps the most convenient and suitable is cadmium, which is enclosed in aluminum or stainless steel sheets. Another possible material is boron in stainless steel, or boron in carbon steel, protected by a plating or cladding. A third possible control material is boral, a matrix of boron carbide in aluminum with aluminum cladding. Still another, which is frequently used as a substitute for hafnium, is an alloy of cadmium-silver.

Various provisions have been made for the control rod guides. In some reactors, the fuel assembly boxes provide the guide surfaces. In others, channels of the core structural material contain the rods. Where full length guide channels are not incorporated in the core, bearings must be provided. These may be made of polyethylene, nylon, Teflon, or metal. The control rods and guides are always made long enough so that the top section is within the topmost guide when the rod is down, and the rods do not pull out of the guides when in the fully withdrawn position.

8.4 CONTROL ROD DRIVES AND POSITION INDICATORS

The design of the control rod drives should take into consideration the purpose of the control rod, i.e., whether it is for safety, for control, or a combination of the two. This will determine whether the rod is normally full in or full out, and whether its position must be adjusted accurately. A rod used exclusively for control would not require a scram mechanism. Critical experiments have been designed and operated with both single-purpose and dual-purpose rods.

The mechanism for withdrawing the control rod may be of several forms. It may be a lead screw device, a rack and pinion, a friction drive, a cable device, or it may be pneumatically or hydraulically operated. The rack and pinion method has certain distinct advantages. The engagement is positive so that rod position indication may be derived from the rotating shaft. The drive can proceed in either direction; that is, the pinion gear can drive the rack or the rack can drive the pinion gear. This allows the engaging mechanism to be located on the rotating shaft. The rack and pinion mechanism has been used on a majority of assemblies, although some have used cable drives. Safety rods must have a quick release mechanism. In critical assemblies this may take the form of an electromagnet which supports the rod, or magnetic clutch which engages the driving gear of a rack and pinion drive. By disconnecting the power to the magnet or clutch, the rod falls freely into the reactor.

The electromagnet type of pick-up, where the rod itself is supported by the magnet, may be used with any of the withdrawal mechanisms mentioned. This type of device has the important advantage that, when disengaged, the rod falls entirely free. There are some significant disadvantages, however. The drive must be driven down to pick up the rod. The rod position indicator is usually attached to the drive and shows the position of the drive rather than the position of the rod. The latter disadvantage can be overcome in part by an indicator which shows when the rod is latched.

The magnetic clutch type of latch can be used with the rack and pinion, the friction, and the cable drives. It has the advantages of providing a drive that can pick up the rod immediately after a scram. The rod position indicators may be located on the rod side of the clutch, giving a continuous indication of rod position. The disadvantages are not serious. The magnetic clutch adds the inertia of half the clutch to the rod inertia, which limits scram speed. However, this rotational inertia is usually very low. During a scram the pinion gear is always engaged, and there is the possibility of the gear binding or of material catching in it. This may be avoided to a large extent by careful mechanical design. The magnetic clutch has been widely used on critical experiment control rod drives.

The pneumatic type of drive may be applied to safety rods, but it has not proved useful for control rods because an initial pressure must be built up to overcome the friction of the piston. This leads to jerky rod motion and can cause difficulty in fine positioning of the rod. There are now commercial units available, combining pneumatic drives with hydraulic damping cylinders, which might eliminate this particular difficulty. Hydraulic cylinders may be used directly as rod drives. The pneumatic and hydraulic drives may be used with the electromagnet pickup or may be connected directly to the rod. In the latter case, the drive would be designed for a fast rod insertion rather than gravity fall. This could be done by using a double acting cylinder with moderate pressure applied to the top section at all times. Rod position would be controlled by varying the pressure in the lower cylinder above or below that of the top cylinder. Scram would be induced by removing the lower cylinder pressure.

Pneumatic power has been used for driving safety rods, but neither pneumatic nor hydraulic power has been commonly used for operating control rods. In most cases, control rods for critical assemblies are powered by direct-current motors with worm gear speed reduction. Worm gears are preferable to spur gears because they are usually stronger, have greater torque ratings, and cannot be driven by the motion of the rod as can spur gears.

Variable speed is usually a desired characteristic of a rod drive to give flexibility of operation and accuracy of positioning. This has been accomplished by the use of direct-current motors with electronic power supplies or selenium rectifier direct-current supplies from a variable voltage transformer.

For safety reasons, the power supply for rod drive motors is designed to operate a single rod. If more than one rod is connected, the power supply is incapable of the necessary output. Thus, the rods are driven at reduced speed, or will not move at all. In this way, the safety limit set on the rate of change of reactivity will not be exceeded.

A shock absorber is required to stop an unlatched rod that is falling freely. Oil-filled automobile shock absorbers have been adapted for this purpose. An oil-filled dashpot designed into the rod drive serves equally well and occupies less space. Dashpots which depend on the moderator water for their action are not desirable because they require water in the assembly for testing. This would complicate safety rules during check out.

The position indicator incorporated into the rod drive unit will depend on the purpose of the particular rod, the design of the reactor, and the information desired from the critical assembly. Although purely mechanical position indicating devices can and have been used in some critical assembly work, the more satisfactory position indicators are remote electrical devices. A-c selsyns, d-c selsyns, linear potentiometers, rotating potentiometers, and multiple indicating lights are all possible position indicators. Of these, a-c selsyns have proved very adaptable and reliable, and have been used on the majority of the rod drives.

A single pair of selsyns (transmitter and indicator) has been used on some drives. In this case the pair is used with a two-hand, geared indicator: One hand indicates the coarse position of the rod, making one or less revolution for the full travel of the rod, while the other hand indicates the fine position, making one revolution for each inch or each centimeter of rod travel. With the single indicating selsyn the units must have sufficient torque to drive the gear train. A single selsyn pair has also been used with a digital indicating device such as a Veeder Root counter. Such a drive is not as desirable because the rod position is not as quickly apparent to the operator as it would be from hand indicators on a dial.

The single pair selsyn system has a drawback in that the transmitting selsyn makes a large number of revolutions for the full rod travel and, depending on the design of the system, the transmitting selsyn or the receiving selsyn may not follow during a rod scram. This leads to the requirement that a method be provided for resetting the zero on the rod position indicators. On a reactor having a large number of rods, this can be an undesirable feature.

The most commonly used indicating system uses two pairs of selsyns to indicate the position of a rod. One pair makes one revolution or less for full rod travel, giving a coarse position indication; the other pair makes one revolution per inch or per centimeter of rod travel, thus giving fine indication. In this case, the fact that the selsyns will not follow a scram is not important (when two-pole selsyns are used), since they return to zero after the rod drop is complete. To date, a fine selsyn indicating to the nearest 0.01 inch has been satisfactory. With higher fuel loading and larger reactivity per unit control rod travel, however, there is a growing need for more accurate rod position indication.

One problem that should be recognized in designing a position indicating system occurs in designs in which a selsyn makes many revolutions during the full rod travel. When the rod is stopped by the shock absorber during a scram, the rotating selsyn will also be stopped

suddenly, subjecting the selsyn shaft to abnormally high torque loads. This may be alleviated by a rubber coupling to the selsyn shaft, or by a magnetic coupling that can slip under high torque.

The stress problem with the selsyn shaft can, in principle, be alleviated by connecting the fine position selsyn, or both selsyns, on the drive side of the magnetic clutch. This solution is not recommended because it raises the new problem of zero of position indication. Further, it removes from the operator knowledge of the rod position following a scram.

For safety rods that are used only in the "in" or "out" position, a coarse indicator alone is satisfactory. For safety reasons it is desirable to know where the safety rod is located. Therefore a safety rod should not be operated without some position indicator built into its drive unit.

8.5 NEUTRON START-UP SOURCE

The neutron source used for approaching criticality must be intense enough to produce counts on the nuclear instrumentation when the reactor is completely shut down. Within this limit the intensity should be as low as possible to reduce the biological hazard to personnel. A source having a low gamma activity, such as polonium-beryllium, is usually used in critical assembly work.

A critical assembly, in contrast to other forms of reactors, is one in which frequent changes are made. New loadings, geometries, and control rod configurations are introduced at frequent intervals. The source is useful not only for start-up but also for safety in loading the core. The neutron leakage, and therefore the multiplication, can be measured continuously during core loading, or it may be measured periodically after each stepwise addition of fuel to the core. While continuous monitoring is to be recommended, it exposes the workers to greater radiation than does the periodic measure of multiplication.

For start-up purposes, if the source is used within the core, one curie of polonium in a polonium-beryllium source has been found satisfactory. This source has a total neutron flux of approximately 2.5×10^6 neutrons per second. Because of the decay of polonium, it is customary to obtain a source approximately 10 curies in strength, and to use it until it is below the one curie level (slightly over 1 year). For critical assemblies, where the source is brought up to the core rather than introduced into the core, the effectiveness of the source is reduced by a factor of 5 to 10 or more, and stronger sources are therefore required. For such applications it has been customary to use sources with initial strengths of 20 to 30 curies. Because these sources do not enter the core, they can be physically larger and, therefore, may be more efficient neutron emitters.

Polonium-beryllium sources are commonly contained in stainless steel or nickel capsules that are sealed by nickel plating. A standard, easily obtainable source is cylindrical in shape, 0.7 inch in diameter and 0.7 inch long. A source of this size cannot be inserted into a reactor core easily, but it is useful for start-up when applied at the surface of the core. Sources may be obtained in special capsules made to customer specifications. A source commonly used for reactor start-up is cylindrical in shape, 0.225 inch in diameter with an over-all length of 1.25 inches. The source proper is in a 1-inch section of this length; the remaining length is a ball-shaped section which permits a ball and socket connection to be made to the source drive mechanism. Such a small source is easily introduced into the reactor for start-up.

A small source is desirable, but there is a lower limit to source size for a given neutron intensity. As sources become smaller, the amount of beryllium present becomes smaller, and the self-absorption of alphas in the polonium increases. Table 8.1 gives a rough indication of how the source efficiency varies with the density of polonium in the source. The year of source manufacture is also given in this table because there is some indication that, in recent years, improved manufacturing techniques have increased the efficiency of sources.

Table 8.1--NEUTRON EFFICIENCIES OF POLONIUM-BERYLLIUM SOURCES

Source Density, Curies of Polonium per cc of Source Volume	Source Efficiency, * percent	Year of Manufacture
13	87	1956
40	80, 85	1956
41	78	1955
40	62	1952
100	30	1952

* 100 percent efficiency is equivalent to 2.85×10^6 neutrons per second per curie of polonium.

8.6 SOURCE DRIVE MECHANISM

In critical assembly work the operating power level is so low that the source used for start-up will have a strong effect on the behavior of the assembly. It is, therefore, necessary to remove the source during critical operation. To obtain the maximum effectiveness from the source neutrons, it is desirable to locate the source within the core, preferably close to the center. As already noted, where it is difficult to find physical space for so locating the source, it is brought to the surface of the core.

If the source is used for reactor start-up only, and is not part of the experiment, it is necessary to provide only coarse position indication for it. Location of the source to within one-half inch may be sufficiently accurate. This precision can be provided by a single pair of selsyns.

A shielded container for withdrawing the source should be provided. This provides protection for personnel working near the reactor when it is shut down, and minimizes the effect of the source on the reactor during operation.

Drive units of the type described for control rods may also be used as source drives. Another possible type is a flexible cable that pushes the source through a tube. The source is always fixed to the drive, since latching and scram devices are not required as with control rods.

The speed at which the source is driven is subject to fewer limitations than is the speed of a control rod drive. If insertion is too rapid, a power rise may result and that may initiate a scram. However, this is not a hazardous situation. The condition can be controlled by intermittent driving of the source, or by use of a variable speed source drive. Both systems have been used. Twelve feet per minute has been found to be a convenient source speed for several critical assemblies.

Limit and interlock switches are incorporated into the source drive. Whenever possible these operate from the source, or the source holder, in order to minimize the possibility of

actuating these switches when the unit malfunctions. The limit switches operate at each end of the source travel; the interlock switch provides greater latitude if it allows the operation of the reactor controls when the source is within a designated region. Typical permissive regions may be within the core or within the central third of the core.

8.7 NUCLEAR INSTRUMENTATION

The instrumentation for a critical assembly may be divided into two general classifications: That used for operation and safety, and that used for data taking. In a given critical experiment the two classes may be entirely separate, or may have considerable overlap.

In the mock-up critical assemblies, large reactivities are usually available. For safe operation of such a facility, one typical set of rules requires four or more operating channels which incorporate a minimum of four power level scrams and two period scrams. Of these, the rules further require that three level scrams and one period scram be in operation whenever the reactor is in operation. This rule provides for some inoperative channels and allows maintenance while the reactor is in operation. Usually, more than the minimum number of scram circuits are effective during operation.

Generally speaking, direct-current detection devices have been preferred over pulse-counting devices for operational and safety channels. In most cases these devices have longer useful ranges and better overload characteristics than pulse-counting devices. Direct-current detectors that have been used include neutron sensitive boron-lined ionization chambers, gamma sensitive high pressure argon chambers, and photomultiplier-scintillation crystal detectors.

Ionization chambers used with critical assemblies have low current outputs. Hence, high-gain amplifiers are required. For this purpose, vibrating reed electrometers may be used. This system gives a linear output requiring scales to be switched frequently as the reactor power level changes during normal operation. Logarithmic amplifiers give power indication through several decades on a single scale, but at the cost of reduced accuracy. Both types of amplifiers are necessary in the critical assembly instrumentation.

The photomultiplier-scintillation crystal detector amplifies the signal within the tube. Therefore, the output of the tube can be connected to a direct-current meter and a relay, to provide a reactor safety channel. This has a limited range of indication. However, such a channel is satisfactory for safety application if checked frequently, and it is extremely simple. The photomultiplier tube can also be used in conjunction with a current regulating circuit. The high voltage to the photomultiplier, which will vary to hold the current constant, then becomes a measure of reactor power. This produces a logarithmic response. However, the logarithmic characteristic is only accurate over approximately 3-1/2 decades. Though below and above this range the decades are compressed, the arrangement is satisfactory as a safety circuit. The principal disadvantage of a photomultiplier tube is its tendency to age and fatigue when used at moderate or high outputs. This does not destroy its value for safety instrumentation, but it does make ionization chambers more attractive.

The reactor period is obtained by differentiating a logarithmic power level signal from a direct-current detector. The period signal may also be obtained from a count-rate meter, where the count-rate signal is first amplified logarithmically. The period circuits are usually adjusted so that the meters read full scale in a 3-second period with the scram set between 5 and 8 seconds. A diode reverse current bypass and other features to prevent damage to the meter are usually included in the period safety circuit.

Count-rate meters operating from pulse-counting detectors have been used for safety and operating instrumentation in the past. They have the disadvantage of requiring greater care in alignment and, in general, more maintenance than direct current circuits. They are useful over a shorter range than are ionization chambers; their usefulness for safety purposes is limited by a tendency to saturate. For these reasons, the pulse-counting detectors have not been used for safety instrumentation on the more recent critical assemblies.

In critical assembly work where reactor powers are low and there is no large build-up of fission products, the power level can normally be measured with either the gamma activity from the reactor or by the neutron activity. The neutron leakage from the reactor is directly proportional to the reactor power and is the more accurate, particularly where transient measurements are being made. However, for most reactor measurements the gamma level has been found equally suitable.

The signal for initiating a reactor scram is obtained from a relay or a relay type meter. The latter has been used extensively because it has been developed to a point where it is as fast acting and as reliable as a relay. Their advantage is that the relation between the power level, or period, and the point at which the reactor will scram is visually displayed.

The channels used for data taking are usually required to be more sensitive to low level signals and more accurate over the entire operating range. For these reasons pulse-counting detectors are generally used.

Pulse counters which have been used are boron trifluoride proportional counters, fission counters, and photomultiplier-scintillation crystal detectors. The BF_3 counters have the advantage of gas multiplication of the signal, providing therefore a larger output pulse. In fission counters the fission fragments give a high degree of ionization which permits the detection of pulses by collecting the ions without gas multiplication. This leads to shorter pulses, allowing higher counting rates without counting loss, and to a lesser dependence on the voltage applied to the tube.

The direct-current radiation detectors are not generally used for data taking. There is some uncertainty as to the accuracy of the logarithmic amplifiers, and this accuracy is difficult to check over the number of decades covered in normal use. While these and other objections to direct-current data-taking channels can be overcome or corrected, it is necessary to balance the cost and effort to obtain the desired accuracy against the expense of additional channels of instrumentation.

For the initial start-up with a new critical assembly there is considerable uncertainty about the counting rate as criticality is approached and how rapdily the detector level will change between the shutdown condition and the near criticality condition. The uncertainty requires that the criticality be approached slowly, and that the reactor power level be known at all times. Under these circumstances, for accurate low level indication, pulse counters are generally used. After one or more start-ups with a given assembly, information will be available as to the readings of the other detectors during the start-up procedure. For these reaons, it is not necessary to use the pulse counters during subsequent start-ups.

Reactors can be operated from panel instruments or galvanometer indicators. With critical assemblies, where measurements are usually made directly on the reactor, it is desirable to have one or more recorders in the instrumentation to aid in the operation, and to provide a continuous past history of the reactor operation. A logarithmic level recorder is

useful for general operation. This is selected to indicate the power level from complete shutdown to maximum power. For maintaining the reactor power level, a linear recorder is more useful because of its expanded scale and greater sensitivity. One or more of each type have generally been incorporated in each critical assembly.

Period recorders have not been used in the Bettis critical instrumentation. Period indication is useful, however, in operation of the reactor, particularly in start-up and while reactor power level is being changed. One or more period meters are often included where they can be easily seen for operational purposes.

Figure 8.1 shows a block diagram of the instrumentation of a typical critical assembly.

The positioning of the nuclear detectors can be handled in a number of ways. Figure 8.2 shows a telescoping instrument thimble which bolts to a flanged opening in the side of the reactor tank. This permits locating the detectors at a constant distance from the surface of any core in the reactor tank. The instrument thimbles can also be suspended in the reflector region from above the reactor tank. For a fixed core size, a fixed size instrument thimble may be bolted to flanged openings in the reactor tank.

8.8 CONTROL INSTRUMENTATION

In addition to the nuclear instrumentation, a critical assembly requires instrumentation to provide continuous information on control rod and source positions, temperatures, and water level. The control rod position and source indication are required for both safety and data purposes. Temperature indications are required primarily for data purposes. Water height instrumentation must be included for operational safety and to provide experimental data when full water height is not used.

One method of measuring and indicating the water height remotely is to measure the differential pressure between a point in the water and the surface of the water. There are several commercial units based on this principle that use a manometer to measure the pressure required to bubble gas through the water. The instruments are accurate to approximately 1-millimeter water height. These instruments are relatively slow in response (they require several seconds to indicate a change in height accurately), but this has not been a handicap in reactor operations.

The manometers used with these differential pressure devices have indicator fluids of several densities to obtain expanded or contracted scales. This permits a choice of precise height indication, or of compactness where accuracy is not a primary requirement. The manometer scales may be marked in any way desired. It is often convenient to make the zero water level coincide with some reference point on the core (such as the bottom of the fuel section), and to have the scale drawn to show negative and positive water heights. On some assemblies these indicators have been used in pairs, one to indicate water height at all times over the full range of water travel; the other to indicate water height more precisely in certain regions, e.g., near the top of the core and in the top reflector region. Pressure switches are available for these indicators that will sound a warning, operate a shut-off valve, or perform other control duties when the water height exceeds the desired limits.

Another method which has been used to measure the height of water in the reactor remotely is an electrical contact method. The resistivity of the moderating water is low enough to permit this type of device to be used even if the moderating water has no material dissolved in it. A point making contact with the surface of the water is used to provide a

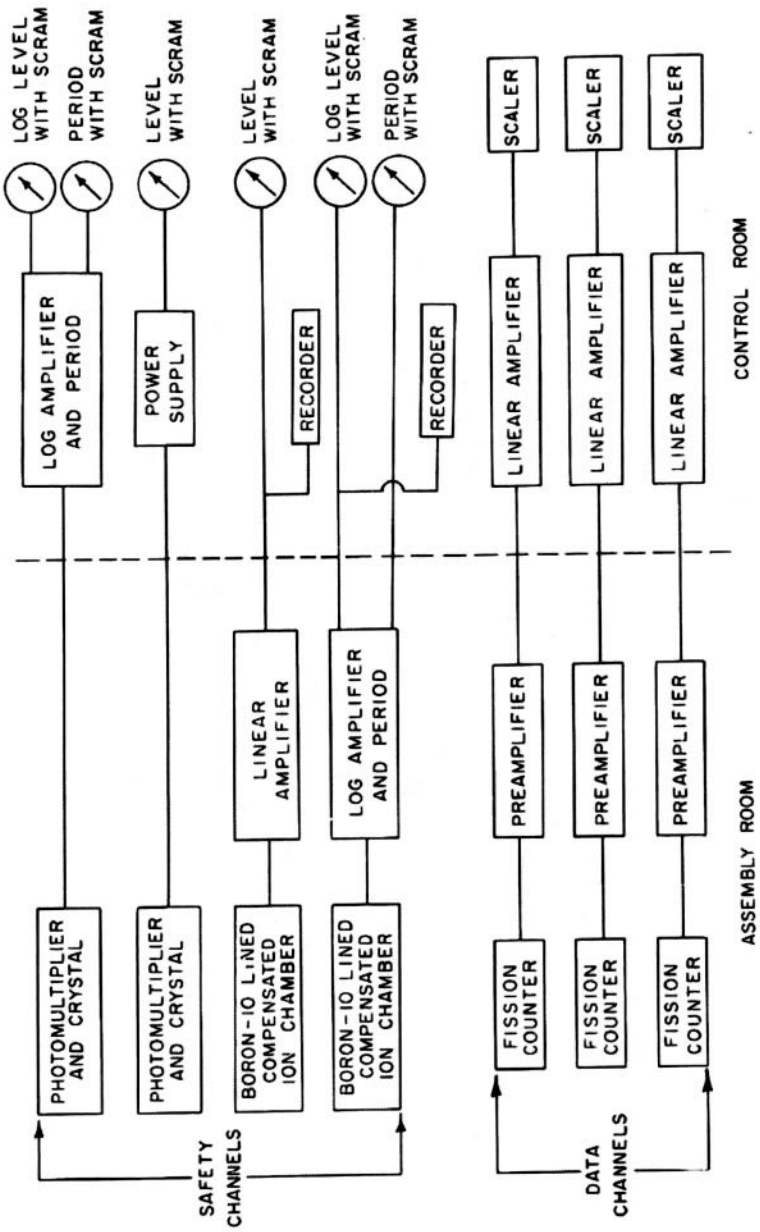


Fig. 8.1 - Block Diagram of Nuclear Instrumentation.

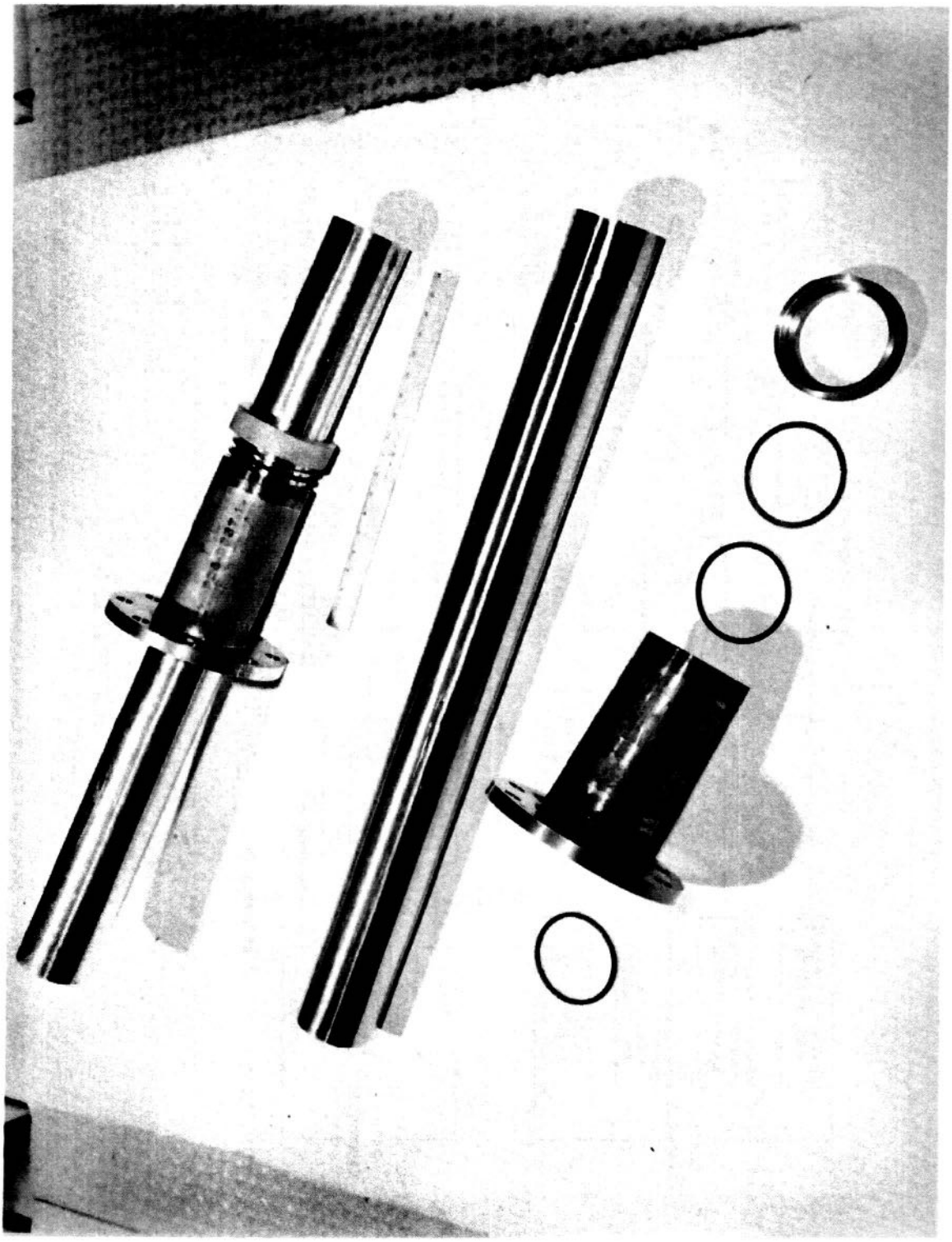


Fig. 8.2 - Telescoping Instrument Thimble (Assembled Thimble Is at the Top, the Components Are Shown at the Bottom).

signal to indicate the height. In a conducting liquid, this method has been used to indicate the surface of the water to a precision of approximately 0.01 inch.

The electrical contact method of determining water height can be used in three different ways: It can be used with fixed points, so that when the water level reaches a predetermined point a signal is provided. It can be used with a contact point that has a remote manual drive, so that the point can be driven up and down until the surface is contacted. Alternately, it can be used with an automatic system in which the point hunts the surface by continuously making and breaking contact. With either of the latter two systems, the fixed points may be incorporated for calibration to check the accuracy of the system.

Another method, adaptable to low conductivity water is a modified electrical contact method. Here a metal float is used with a constraint, such as a guide tube or guide wires. The probe has two contacts which, upon touching the surface of the metal float, complete the electrical circuit to provide an indication of the float position and, therefore, the water height.

Instrumentation is also provided in critical assemblies to indicate the level in all storage tanks. This permits a check on the system during start-up and operation. For example, if the water is not rising in the reactor tank at the expected rate, a check on the rate at which the storage tank level is dropping will show whether the difficulty is due to malfunctioning of the pump or valves, or whether it is due to a leak in the system.

The accuracy required of temperature instrumentation depends on the experiments to be performed. For most reactor operations, an accuracy of ± 0.03°C, which permits a direct reading accurate to the nearest tenth of a degree, is adequate for temperature coefficient studies. Reactor temperatures have been measured with platinum-resistance thermometers and with thermistors. The platinum-resistance thermometers are used in bridge circuits, either manually operated or recording. The platinum-resistance thermometers are nearly linear with temperature and, in one case, were compensated by a resistor network so that the reading in ohms was equal to the temperatures in degrees over a limited range. Thermistors have a more rapidly changing resistance with temperature changes, and therefore are more sensitive. They are not linear, and have been found to drift with age (approximately 1 C° per year). Thermistors must be calibrated individually, and the calibration repeated periodically.

8.9 ELECTRICAL INTERLOCKS

A few simple basic electrical interlocks are included in each critical assembly control system to ensure the proper sequence of operations. Before a control rod can be raised, or before water can be added to the core, one such interlock requires either that (1) the neutron start-up source be in the reactor or (2) the neutron flux be above some arbitrarily set measurable value. Some of the critical assemblies are further interlocked, so that if the above situation is not satisfied a scram occurs. In most assemblies, however, failure to satisfy the so-called flux-up condition simply prevents a positive reactivity change from being made.

Another interlock, basic to all critical assemblies, requires a rod to be withdrawn above the core before water may be added to the reactor tank. Because of the possibility of a control rod jamming, typical operating rules have actually required that two rods be withdrawn, one completely withdrawn, and the other either partially withdrawn or fully withdrawn at the option

of the operator. More recently, an interlock system has been designed which requires two rods to be withdrawn, and this has been incorporated in certain reactor consoles.

A third interlock is included to limit the rate of addition of reactivity to the core. This makes it impossible to raise a control rod and to add water to the reactor tank simultaneously. The critical assemblies have in some cases been designed so that the rod motion would override any pumping operations.

In addition to these operational interlocks, the assembly room door is interlocked to produce a scram if it is opened after reactor operation has started.

8.10 MANUAL SCRAM

The instrumentation of a critical assembly includes a manual scram switch to interrupt the holding current to the clutches or magnets and allows the control rods to drop. The switch also interrupts the power to the main dump valve, allowing water to return to the storage tank. This results in a full reactor scram. The switch is so wired that once a scram has been initiated, releasing the switch does not interrupt the scram. It is necessary to reset the switch either manually or electrically, or both, before the reactor can be put into operation again. In some reactor assemblies, additional partial scram switches have been included which will only dump the water or only drop the rods. These are convenient but not necessary for safety. The rod drop switch is sometimes used to conduct an experiment.

8.11 CORE SUPPORT STRUCTURE

The core support structure is that unit, or assembly of units, which supports the core itself in the critical assembly. This may be a grid or table device in the bottom of the reactor tank upon which the core is placed, with a corresponding plate or grid to position the top of the core. Alternately it may be a support plate above the reactor tank from which the core is suspended. Typical designs are shown in the descriptions of specific reactors in Chap. 9.

The principal mechanical requirement is that it must have sufficient strength with minimum deflections under load. It must position the core accurately and hold it securely. Further requirements are that it must permit the desired flow through or around the core support. It must permit the desired access to the core, such as may be required for changing the loading, or for inserting or removing flux measuring foils. An additional requirement may be that it permit the modification of the core geometry. This may be accomplished by replacing the entire support plates or by building part of the support structure of small, removable units. Both methods have been used.

8.12 ASSEMBLY SUPPORT PLATFORM

To permit draining by gravity in a water moderated critical assembly the reactor tank, control mechanisms, and associated equipment are mounted at some distance above floor level. A structure is required to support these units in position. In the design of such structures, strength and deflection under load are both important. Often the limitation on deflections dictates the choice of structural sizes. A further requirement for the support structure is accessibility. Large, heavy structural steel shapes are preferred to small sections with a large amount of cross brazing. A platform is required to provide working convenience, with sufficient head room for core assembly work, insertion and removal of foils, and for maintenance of the electrical and piping systems.

The type of assembly platform used has become more or less standardized. It is a two or three-level platform supported by heavy columns, and generally equipped with two reactor tanks. Somewhat different types of support structures have advantages of improved accessibility. The assembly stand used for the STR-CA, which is constructed principally of pipe and supported by two large columns, is a nonstandard type having excellent accessibility, particularly for shielding studies. Approximately 160 degrees of the core periphery are available in this design.

A most desirable feature of the support platform is that it be designed to provide space for the accessory equipment associated with the critical facility operation. This may include shelves for electronic equipment, test locations for rod drive units, and storage racks for spare units such as control rods and control rod drives.

8.13 WATER-HANDLING SYSTEMS

The water-handling system includes a reactor tank, one or more storage tanks, filling pumps, control valves, and associated piping. In addition, there may be a heating system consisting of a heat exchanger, circulating pumps, control valves, and the connecting piping.

Reactor tank design depends on several factors. One is the size of the cores to be accommodated. Allowance must be made for a bottom and top reflector; this space also permits improved circulation of water in temperature coefficient studies. If not governed by other considerations, a radial reflector thickness of 6 to 8 inches provides room for insertion of equipment such as detectors, complete reflection for the core, and a moderate amount of shielding to the operating staff. In the design, consideration should also be given to the fluids that may be used in the system, i. e. whether water only, or solutions and other hydrogenous liquids.

Fabrication, rather than material, accounts for most of the cost of the reactor tank. For this reason, stainless steel has been preferred to aluminum. Stainless steel has the advantage of easy maintenance and corrosion resistance. Tanks fabricated from plastics, or from Fiberglas reinforced plastics, are possibilities that have been used at other laboratories.

The top of the reactor tank has a flange with a bolt circle. This stiffens the tank, and also provides means for lifting and for attaching equipment to it. If the tank is to support the reactor core itself, or important associated equipment, it must be leveled. The tank must have locations for the nuclear detectors. These consist of flanged openings in the side or bottom to which instrument thimbles may be attached. The bottom may be integral with the sides or may be separate, serving a double purpose of tank bottom and tank support.

A dump-line connection, from 4 to 8 inches in diameter, is built into the tank at, or near, the bottom. The filling line connection of most tanks is located near the bottom to avoid splashing. This connection is often incorporated into the dump line just above the dump valve. When a heating system is associated with the facility, connections for the water circulating line are provided in the tank.

Special features that may be required by a particular facility are incorporated into the reactor tank. For example, a special thimble for a start-up source insertion may be provided. Another feature might be a well for dropping fuel-bearing control rods out of the core.

The storage tanks required for critical assemblies are usually less intricate than the reactor tank, and standard sizes may be used. Here again, stainless steel is more dependable, and is to be recommended.

The piping system is constructed of material that will not corrode significantly. Stainless steel, plastic, and aluminum have been found satisfactory in most applications. The piping system is designed to be self-draining with minimum or no holdup. The piping may be assembled in place by welding, or by using any of the many commercial fittings available. In a number of cases, the piping was welded in sections, with flange connections used to join the various sections. This allows most of the assembly work and the preliminary cleaning to take place away from the reactor, and permits easy disassembly for future cleaning.

For pumping water and similar liquids, the centrifugal pump is used. Since centrifugal pumps cannot create high pressure, there is no need for relief valves and bypass lines. This simplifies the piping layout. It is also possible to drain liquids back through the centrifugal pump, which further simplifies the piping system. A difficulty with centrifugal pumps is that a poor shaft seal will introduce air bubbles into the system. For some applications, such as for the spray nozzles on the DCTF, the use of positive displacement pumps to produce higher pressures is required.

The Grinnel-Saunders type diaphragm valve has been chosen for most applications on the critical assembly system. It has no packing or seals to leak, offers little resistance to flow, is tight closing, and can be positioned for complete drainage with no holdup of water or solution. This valve is available in a large range of pipe diameters, in a large variety of pipe connections, and in many types of body and diaphragm materials. It is also available with a variety of actuators, such as manual, air, electric solenoid or electric motor. Quick acting or slow acting types can be procured.

The Grinnel-Saunders type of diaphragm valve with an air actuator has been used in a larger size as a dump valve because it performs well and is available commercially. However, it is bulky, heavy and expensive. One other type of dump valve, the flap type, has also been used. In this design a cable pulls a flap up tight against the end of a pipe to prevent escape of the water or liquid. The cable is held tight by a magnet or a magnetic clutch. On a proper signal, the power to the magnet is switched off allowing the flap to open.

Remotely operated valves are obtainable either normally opened or normally closed. The choice for a particular application is determined by safety considerations. The dump valve is normally open, so that a power failure causes a reactor shutdown through the loss of moderator. The filling valve is usually normally closed, so that in case of failure the filling ceases. The use of remotely operated valves is limited to necessary operational functions of the reactor itself. For associated equipment, such as transfer lines between storage tanks, hand-operated valves are generally used to limit the possibility of operational error.

In the design and construction of a facility, an attempt is made to isolate the various plumbing systems whenever possible. For example, the heating and circulating system might be isolated from the storage and filling system. Such a separation of functions may take extra pipe, extra valves, and extra pumps, but there is a gain in safety. Where systems cannot be isolated, the design and layout are made such that a minimum of two personnel errors or equipment failures or a combination are necessary to create an unsafe condition.

The reactor heating system for temperature coefficient studies is determined by the size and nature of facilities, and by the amount of high temperature work to be done. Electrical immersion heaters are convenient where the system volume is small (under 1000 gallons), and where the amount of high temperature work to be done is limited. The immersion heaters are installed along the edge of the tank in the reflector, and the water circulated by natural

convection or by propeller stirrers. This method requires no additional plumbing. However, electrical heaters in the larger tanks are slow and require large amounts of power.

Another method of supplying heat to the system is with steam. For one critical assembly, this method was applied by placing a steam jacket around the outside of the tank. With others, it was found convenient and simple to use an external heating system in which reactor water was circulated through the heat exchangers which, in turn, were heated by steam.

The mixing of water to obtain a uniform temperature throughout the reactor is a serious problem. Mixing is aided by the use of stirrers in the tank and by the use of circulating pumps. Deflectors and baffles within the reactor tank to guide the flow may aid in giving more rapid mixing. Mixing may be a particularly serious problem in reactors where there is no cross flow.

8.14 BUILDING DESIGN

Since this discussion is primarily concerned with reactor design, only a brief comment will be made on the building features required for critical assembly work.

The critical assembly itself requires an assembly room, a control room, and a vault for storing fissionable material. A room arrangement which has been found especially suitable to critical assembly operations is to place the vault on the first floor and a control room directly above it, with a connecting stair well along one side of the assembly room. These three rooms, with the stair well, form a unit which permits the critical assembly area to be shut off from the remainder of the building.

A feature common to most buildings used for water-moderated critical assemblies is the considerable height of the assembly rooms. Water-moderated critical assemblies make use of gravity drain of the moderator water as a shutdown safety mechanism. This requires the reactor tank to be at an elevated location or the storage tanks to be below floor level. Control rods are made vertical so that water seals are not necessary, and so that gravity may be used for quick shutdowns. The control rod drive units must have a height somewhat greater than the length of the control rods in order to allow the rods to be fully withdrawn. Most assembly rooms for water-moderated critical assemblies are over 30 feet high, and a 50-foot height is not uncommon.

Critical facilities are made in the open part of the room, so that they may be easily constructed and modified, and so that measurements may be taken easily. Shielding is provided by the walls of the room. These may be thicker than the normal building walls. From 2 to 5 feet of concrete are commonly used for shielding. This can be on one side or on all four sides of the room, depending upon the protection needed.

BASIC OPERATIONAL EXPERIMENTS USING WATER MODERATED CRITICAL ASSEMBLIES

by F. J. Jankowski

8.15 APPROACH TO CRITICALITY

For safe reactor start-up the external source of neutrons, described in Sec. 8.5, is inserted into the reactor core. An approximate equation relating ϕ, the volume average thermal flux in the reactor, and S, the total neutron source strength, is given by Eq. (8.1):

$$\phi = \frac{S}{V\Sigma_a(1 + B^2M^2)(1 - k_{eff})} \qquad \text{Eq. (8.1)}$$

In this equation Σ_a is the total macroscopic absorbtion cross section for thermal neutrons, V is the volume of the reactor, B^2 is the geometrical buckling of the reactor, and M^2 is the migration area for fission energy neutrons in the reactor. The effective multiplication constant of the reactor k_{eff} is related to the infinite multiplication constant k_∞ by the customary equation

$$k_{eff} = \frac{k_\infty}{1 + B^2M^2}$$

Equation (8.1) is derived directly from the two-group diffusion equations by making several assumptions. First, it is assumed that the source neutrons are distribution uniformly throughout the reactor. Second, it is assumed that the source neutrons are emitted at the same energy as fission neutrons. Third, it is assumed that the flux distribution in the reactor is the fundamental mode of the equivalent bare reactor, i.e.,

$$\nabla^2\phi + B^2\phi = 0$$

for both the fast group flux and the thermal group flux.

Despite its approximate nature, Eq. (8.1) is extremely useful. From it we observe that the average flux in the reactor is proportional to

$$\frac{1}{1 - k_{eff}}$$

Now if we consider the reactor to be subcritical, and that it is being modified in some regular fashion to bring it to criticality, a quantity proportional to the reciprocal of $\overline{\phi}$ will approach zero. This is the basis of the usual criticality plot used to predict the critical condition before it is reached.

In the usual criticality approach plot the abscissa is a scale of the change being made in the reactor to achieve criticality. This change might be control rod position, reactor size, water height, fuel loading, etc. The ordinate is proportional to the reciprocal flux in the reactor. The abscissa at which the reciprocal flux curve is zero is the predicted critical condition. To apply this scheme for controlling the approach to criticality, the relative flux level in the reactor is measured by a neutron sensitive detector as criticality is approached. If C is the count rate of the detector,

$$\frac{1}{C} = A_o(1 - k_{eff}) \qquad \text{Eq. (8.2)}$$

where A_o is a constant of proportionality.

Equation (8.2) predicts that the approach curve in k_{eff} is a straight line, with criticality at the point at which 1/C is zero. It is implicit in the derivation of Eq. (8.2) that the count rate measured is proportional to the volume average flux in the reactor. If the neutron sensitive detector is located inside the reactor, this is approximately correct. If, however, the neutron sensitive detector is outside the reactor, its count rate depends upon the neutron leakage from the reactor which may or may not be proportional to the volume average flux in the reactor. Thus, as criticality is approached, the curve may be concave or convex downward, depending upon the geometry of the arrangement.

In a given experiment, a practical method of making the first approach to criticality is to use a control rod to supplement the principal variable. In this system two approach curves are taken: One with the control rod in; the other with the control rod out. Each step of the change toward criticality is made with the control rod in, and is of such magnitude that the counting rate with the change made is less than the counting rate prior to the change with the control rod withdrawn. This assures the operator that the steps being taken in approaching criticality are smaller in reactivity than the worth of the control rod between its two positions.

As criticality is reached, the counting rate becomes extremely large, or does not level out following the change. At this condition the source should be withdrawn. If the reactor power remains level, or drops a small amount and remains level, the reactor is critical. If the power level drops, the reactor is subcritical. The operator must feel his way carefully in the last portion of the approach to criticality. He can obtain a rough estimate of the reactivity remaining to achieve criticality by noting the amount of the initial power drop when the source is withdrawn. This drop can be estimated by extrapolating back to the time of the source removal on one of the power level recorder charts. The theory is described in Sec. 8.17c. Even though the source motion may be too slow to apply the source jerk theory, this scheme gives an approximate value of the δk required to achieve criticality.

8.16 DETERMINATION OF THE POWER LEVEL

The determination of the power level of a critical assembly may be important for various reasons. Depending upon the need, the power level may be obtained approximately or with some precision. There are three possible ways of obtaining a power determination of a critical assembly without operating it at high powers which would raise the temperature of the assembly: (a) Operating the reactor at critical with a neutron source introduced into the core; (b) operating the reactor subcritical with the source; and (c) using foil activation.

(a) Critical Reactor with a Neutron Source. The first procedure requires the use of a neutron source, which may be the reactor start-up source. Care must be taken that the source

(or its holder) does not absorb neutrons so strongly that it has a significant control rod effect. For the simplest interpretation the source should be used at the position of maximum effectiveness. If the source is used at other locations, such as the surface of the core, some experimental data should be available for estimating the effectiveness at that position.

An approximate theory of this method can be developed rather simply. If the reactor is just critical without the source, and a source of neutrons is introduced into the reactor, the source neutrons will be multiplied continuously and the power of the reactor will rise continuously. The fractional power rise of the reactor per unit time will drop off as the reactor power level rises. The power level at a given time can be deduced from the time required for the power level to rise by a given factor, for example, to double.

Suppose that the source introduces S neutrons per second. Each of these neutrons which produces a fission will add to the fission rate of the reactor by $\frac{1}{\ell}$ where ℓ is the mean lifetime of fission neutrons in the reactor. The argument is that the reactor is just critical without the source, so that each fission just maintains itself and leads to $\frac{1}{\ell}$ fissions per second. Of the source neutrons the fraction

$$\frac{1}{(1 + B^2 \tau_s)}$$

reaches thermal energy (with resonance capture not present) where τ_s is the age of the source neutrons if the source were distributed geometrically just proportional to the fission rate of the principal mode of the reactor, and B^2 is the buckling of the principal mode of the reactor. Since the source is concentrated, it is more effective by a weighting factor W. Therefore

$$\frac{WS}{(1 + B^2 \tau_s)}$$

of the source neutrons reaches thermal energy per second. Of this the fraction $\left(\frac{1}{1 + B^2 L^2}\right)$ where L is the thermal diffusion length, are captured in the reactor and the fraction f, where f is the thermal utilization of the fissionable isotope, is absorbed in fuel and the fraction $\left(\frac{1}{1+\alpha}\right)$ produces fission, where α is the ratio of radiative capture to fission capture in the fuel.

Thus the fission rate of the reactor increases by

$$\frac{WSf}{(1 + B^2 \tau_s)(1 + B^2 L^2)(1 + \alpha)\ell}$$

fissions per second per second. Taking 3.1×10^{10} as the number of fissions per joule, the power level of the reactor changes by

$$\frac{dP}{dt} = \frac{1}{3.1 \times 10^{10}} \frac{WSf}{(1 + B^2 \tau_s)(1 + B^2 L^2)(1 + \alpha)\ell} \text{ watts/second} \qquad \text{Eq. (8.3)}$$

In applying this method, the absorption of the source drive equipment may be corrected experimentally by calibration of the drive without the source against a control rod. A detailed derivation has been given by Garabedian.[1]

(b) Subcritical Reactor with a Neutron Source. A power determination may also be made by inserting the neutron source in the core when the core is subcritical by some definite, measured amount. The measured subcriticality can be found in a separate experiment, for example, by a rod calibration. Let $1 - k_{eff} = \Delta k$ be the amount of this subcriticality. If, again, S is the number of neutrons per second emitted by the source, S will be multiplied by

the quantity $1/(1 - k_{eff})$ or $1/\Delta k$. Then the number of fissions per second will be

$$\frac{WSf}{\Delta k(1 + B^2\tau)\,(1 + B^2 L^2)\,(1 + \alpha)}$$

where τ is now the age of fission neutrons. The power level corresponding to the fission rate will be

$$\frac{WSf}{\Delta k(1 + B^2\tau)\,(1 + B^2 L^2)\,(1 + \alpha)}\;\frac{1}{3.1 \times 10^{10}} \qquad \text{Eq. (8.4)}$$

This power level can then be related to the critical assembly detector readings.

(c) Foil Activation. Foil-activation techniques may also be used for power determinations. The foils are used as a means of determining the flux which, in turn, can be multiplied by the fuel density and fission cross section, and integrated over the reactor volume. If the fuel density is constant, it is only necessary to average the flux over the reactor. If the geometry of the reactor is simple, the flux distribution may be calculated, and it is necessary to check the flux with foils at only a few points in order to calculate the average flux. With this method, calibrated foils and counters must be used to obtain the absolute activity in the foils. If uranium foils can be used that have the same enrichment as the fuel in the reactor, it is possible to obtain the power more directly without many of the corrections required with other type foils. In this method, the accuracy of the determination will be strongly dependent on the accuracy of the absolute determination of the flux, and on the accuracy with which the flux distribution is known. For a discussion of activation and counting procedures, see Chap. 10.

8.17 CALIBRATION OF CONTROL RODS

There are many experimental methods for calibrating control rods. These include the rod-bump period method, in which the data are interpreted by use of the inhour equation, the subcritical multiplication method, the source jerk method, the rod oscillation method, and the rod drop method. Some of these methods give the differential value of the control rod worth. To extend rod calibration, other changes in the reactor core (in the position of other control rods, or in fuel or poison loading) must be made so that a new position of the rod under calibration is possible. Because of these changes, a rod calibration is significant only for the particular conditions under which it is obtained.

(a) Rod-Bump Period Method. The rod-bump period method of rod calibration has often been used because it can be applied to any reactor in which the control rod can be moved and the motion measured. The only instrumentation required is means of measuring neutron level to the accuracy desired. Since this method determines the differential rod worth at a point, the disadvantage is lengthy procedure. This is not a disadvantage, however, if the worth is required as a function of position.

If, in a reactor, the reactivity is positive and constant, the reactor power, and therefore the neutron level, will rise asymptotically on an exponential given by

$$P = P_o \exp(t/\tau) \qquad \text{Eq. (8.5)}$$

where

P = reactor power at time t,

P_o = initial reactor power (at time t = 0),

τ = reactor period.

This equation does not hold immediately after the reactivity change is introduced, but holds after the initial transients have died out. The period is found by differentiating this equation logarithmically with the respect to time.

$$\frac{1}{P}\frac{dP}{dt} = \frac{1}{\tau} \qquad \text{Eq. (8.6)}$$

The period is determined experimentally by measuring the power (or neutron level) as a function of time and from this calculating the reactor period. The reactivity is then determined by applying the inhour equation.[9]

The rod calibration is started by making the reactor critical in the desired initial configuration. For example, in the initial configuration the rod being calibrated may be full in, the compensating rod full out, and all other rods at equal height in the core. Although it is desirable to start with the reactor power exactly level, it is often expedient to accept a long, positive period and measure the period rather than attempting to make the power exactly level. Once the reactor power is leveled, or the long positive period is measured, the rod being calibrated is moved a short distance, or bumped, and after the transients are allowed to die out, the faster reactor period is measured. Approximately 2 minutes are usually sufficient for the transients to die out and the period to be established. The reactor periods may be determined by taking a number of equally spaced and equally timed counts on pulse-counting detectors, such as BF_3 counters or fission counters. Five to seven counts are desirable for a good determination. From the period measurement, reactivity is inferred through the inhour equation. The difference in reactivities between the two rod positions divided by the distance of rod motion, gives the reactivity per unit motion of the rod. This value is assigned to a point within the interval of motion, usually the mid-point.

When the reactor period has been determined, the compensating rod is inserted to reduce the reactor power. The rod being calibrated is then raised to the next point at which the calibration is desired. The compensating rod is moved, as necessary, to level the reactor power or to yield a long, positive period. The entire cycle is repeated as many times as necessary to obtain the desired number of points on the calibration curve. In this procedure it is clear that with little additional effort the compensating rod itself may be calibrated. Another variation is to use a third rod over a fixed distance to give the period, and to move the pair of rods being calibrated just enough to compensate for the motion of the third rod both in and out.

The single rod may be compensated by a bank of rods rather than another single rod, thus obtaining both calibrations. The range over which the bank may be calibrated is quite limited; and it is generally necessary to make other changes in the reactor to obtain a bank calibration over an extended range. For example, fuel or poison may be added to the reactor or its dimensions may be changed. Changes of this nature to the core actually produce a new reactor. Whether the results obtained from such a series of different reactors can apply to the initial condition is open to question.

In using the rod-bump period method, the data for determining the reactor period are taken on two or more detectors, so that if one of the detectors or its accompanying electronic circuits is faulty, the difficulty will be apparent. The rod motion is usually chosen to give periods in the range of 90 to 150 seconds. These periods are easily controlled, allow the taking of data over a sufficient time to give a number of points for determining the period, and are fast enough to give good accuracy in the determination of the period. If, instead of making

the reactor power exactly level a long positive period is measured, periods should be greater than 1000 seconds. Actually, such periods often are in the neighborhood of 10,000 seconds. These periods contribute significantly to the reactivity and should be measured; it should not be assumed that the power is level when it is not. The level at which a period measurement is started should be sufficiently high so that counting statistics are not a source of serious error. The level must be sufficiently low, so that at the end of the run the counting losses are not serious, and the reactor power level is still within the permissible operating range of the safety channels.

There has been no systematic investigation of the errors that enter into this rod calibration procedure. In typical applications, the reactor period is measured with two or more detection channels. If the spread in the values of the period obtained is greater than 3 per cent, the point is repeated. Usually, the values check more closely than this. The accuracy with which the rod position indicators can be read introduces a further error. This error can be as great as 5 per cent but is normally of the order of 1 per cent. Considering the accuracy with which data are taken, the scatter and the reproducibility of the results, it is believed that individual rod calibrations, both incremental worth and integrated values, are accurate to about 2 to 3 per cent. Bank calibration curves derived from individual rod calibrations include the error of the individual rod calibration as well as error introduced by the uncertainty of the bank position. The result is greater scatter in the experimental points for bank calibrations. The calibration curves for banks of rods are believed to have an error of 4 to 5 per cent. These error estimates do not include any error associated with reduction of period data to reactivity.

(b) Subcritical Multiplication Method. The calibration curve for a bank of rods may be extended by means of subcritical multiplication measurements. As the accuracy of this method is not known, it can be judged only by its fit to data taken by other means. Comparisons have led to an estimate of about 10 per cent uncertainty in this type of measurement. This method is valuable because it will give data when no other information or means of obtaining information is available. Further, the method is relatively rapid.

In applying this method, a preliminary step is required. The reactor is first made critical. Next, the reactor is made subcritical by a known small amount, by inserting a previously calibrated control rod. Then the neutron source is introduced and detector readings are taken to determine the relative neutron flux in the reactor. This procedure is repeated for several positions of the calibrated control rod. A plot is made of the inverse counting rate against $(1 - k_{eff})$. This plot is extrapolated linearly over the range required to calibrate the rods of interest. This completes the preliminary procedure.

Starting from the same initial critical condition the rod, or bank of rods, to be calibrated are inserted in a number of steps. The neutron source is introduced and the detector readings are taken at each step. The value of k_{eff} is obtained from the extrapolated curve drawn from the calibrated rod data. This leads to an integral curve. The slope of this curve gives the incremental reactivity worth curve for the bank of rods.

(c) Source Jerk Method. In the source jerk method of calibrating control rods, the procedure is to use the source jerk to determine the degree of subcriticality of the reactor. This is used, as the rod bump is used, to transfer a reactivity worth to a measured control rod motion. The theory of the source jerk procedure to measure the degree of subcriticality will be considered first.

The theory of the source jerk method of determining subcriticality can be derived simply. Starting with Eq. (8.1)

$$\overline{\phi} = \frac{A_1 S}{(1 - k_{eff})} \qquad \text{Eq. (8.1)}$$

where A_1 is a constant representing

$$\frac{1}{V\Sigma_a(1 + B^2M^2)}$$

we note that

$$k_{eff} = C\nu$$

where C is a constant and ν is the number of neutrons emitted per fission.

Now

$$\nu = \nu_P + \nu_D$$

where ν_P is the number of neutrons emitted promptly and ν_D is the average number of delayed neutrons. Defining

$$k_P = C\nu_P$$

where C is the same constant as in the expression for k_{eff}, we have

$$k_{eff}(1 - \beta) = k_P \qquad \text{Eq. (8.7)}$$

where

$\beta = \frac{\nu_D}{\nu}$ is the delayed neutron fraction.

Substituting Eq. (8.7) in Eq. (8.1), and rearranging, yields

$$\phi = \frac{A_1 S}{(1 - k_P)}\left(1 + \beta \frac{k_{eff}}{1 - k_{eff}}\right) \qquad \text{Eq. (8.8)}$$

The term in Eq. (8.8)

$$\left(\frac{A_1 S}{1 - k_P}\right)$$

may be interpreted as the flux resulting from prompt multiplication of the source neutrons while the term

$$\frac{A_1 S}{(1 - k_P)}\beta\left(\frac{k_{eff}}{1 - k_{eff}}\right)$$

may be interpreted as the flux resulting from multiplication of the delayed neutrons.

If, when the reactor is subcritical with the source present and the flux stationary with time, the source is suddenly removed, the flux will drop to the value representing the contribution of the delayed neutrons. If C_1 is the detector response before the source is removed, and C_2 is the response immediately after the source is removed, then

$$\frac{C_1 - C_2}{C_2} = \frac{1 - k_{eff}}{\beta\, k_{eff}} = \frac{\Delta k}{\beta(1 - \Delta k)} \qquad \text{Eq. (8.9)}$$

To apply this method for calibration of a control rod, the reactor is made critical. The control rod is inserted a given distance and the counting rate C_1 is obtained after the source is inserted and the flux level reaches equilibrium. The source is removed rapidly and the neutron level is measured as a function of time. This is extrapolated back to zero time to obtain the value of C_2 from which Δk is computed.

By starting from the same critical position and measuring the subcriticality for several rod positions, an integral rod calibration curve will be obtained. If each measurement is started from a new critical position and the reactivity change is divided by the amount of rod insertion a differential, or incremental, rod worth calibration curve may be obtained.

In principle, this method can be used to measure the worth of banks of rods as well as individual rods, and to measure the shutdown of the reactor.

The principal disadvantage of the method is that the counting rates tend to be low and, therefore, counting statistics are poor. For Δk values of sizable magnitude, the basic premise that the flux in the reactor is distributed in the principal mode is questionable.

Special equipment may be necessary for this experiment. Rapid writing recorders and d-c devices connected to sensitive amplifiers are generally used. One system uses a boron-lined ionization chamber feeding through a Beckman logarithmic amplifier into a Sanborn recorder. Another method of collecting data is to connect the output of a scaler to a recorder with a rapid chart speed, such as a Sanborn or Brush recorder, so that each 1,000 counts or each 100 counts will produce a mark in the chart trace. From these data, the counting rate as a function of time may be computed and plotted.

(d) The Rod Oscillator Method. The periodic insertion and removal of absorbing material in a reactor will result in the power level fluctuating with the same period. Small power oscillations will be directly proportional to the reactivity worth of the material at the location into which it is being inserted. This principle may be used for obtaining an incremental calibration curve for a control rod. The theory and application of this method has been described by Jankowski, Klein, and Miller.[2] A discussion is also presented in Volume III of this Handbook.

The theory of the rod oscillation method will be developed for a one group model with no leakage, in which reactivity only is a function of time. For this model the infinite multiplication constant is equal to k_{eff}. The reactor kinetics equations are:

$$\frac{\partial n}{\partial t} = \frac{\rho}{\ell} n - \frac{\eta f \beta n}{\ell} + \sum_i \frac{c_i}{\tau_i} \qquad \text{Eq. (8.10)}$$

$$\frac{\partial c_i}{\partial t} = -\frac{c_i}{\tau_i} + \frac{\eta f \beta_i}{\ell} n \qquad \text{Eq. (8.11)}$$

where

β is the total delayed neutron fraction; β_i is the delayed neutron fraction of the i^{th} group

n is the neutron density,

ρ is the reactivity [$\rho = (k_{eff} - 1)/k_{eff}$],

ℓ is the mean lifetime of neutrons in the reactor,

c_i is the number density of the precursors of the i^{th} group of delayed neutrons,

τ_i is the mean life of i^{th} group of delayed neutrons,

η and f are the characteristic parameters of the reactor material.

The reactor kinetics equations will now be solved for the small signal approximation.

Write

$$n = n_o + n_1(t) \qquad \text{Eq. (8.12)}$$

and for the i^{th} delayed group

$$c_i = c_{io} + c_{i1}(t) \qquad \text{Eq. (8.13)}$$

where n_o represents the time average neutron density and c_{io} the time average number density of the i^{th} group of precursors.

$$c_{io} = \frac{\eta f \beta_i \tau_i}{\ell} n_o$$

Substituting these expressions in Eqs. (8.10) and (8.11) and neglecting the products of small quantities $\rho(t)\, n_1(t)$, gives

$$\frac{\partial n_1}{\partial t} = \frac{\rho(t)}{\ell} n_o - \frac{\eta f \beta}{\ell} n_1 + \sum_i \frac{c_{i1}}{\tau_i} \qquad \text{Eq. (8.14)}$$

and

$$\frac{\partial c_{i1}}{\partial t} = -\frac{c_{i1}}{\tau_i} + \frac{\eta f \beta_i}{\ell} n_1 \qquad \text{Eq. (8.15)}$$

Equations (8.14) and (8.15) are solved by taking their Laplace transform and solving for $n^*(s)$, the Laplace transform of the neutron density.

$$n_1^*(s) = \frac{\frac{n_o}{\ell}\rho^*(s)}{Z(s)} + \frac{n_1(0) + \sum_i \frac{c_{i1}(0)}{1 + s\tau_i}}{Z(s)} \qquad \text{Eq. (8.16)}$$

where Z(s) is, by analogy with electric circuit theory, considered to be the natural impedance of the reactor. The first term gives the effect of the forced vibration. Taking the inverse transform of the first term, and treating the case where $\rho(t)$ is a periodic function $[\rho(t) = \delta e^{i\omega t}]$, yields for the steady state solution

$$n_1(t) = \frac{n_o \delta}{\ell}\left[\frac{e^{i\omega t}}{Z(i\omega)} - \frac{1}{i\omega Z'(0)}\right] \qquad \text{Eq. (8.17)}$$

where

$$Z(i\omega) = i\omega\left[1 + \frac{\eta f}{\ell}\sum_i \frac{\beta_i}{i\omega + 1/\tau_i}\right]$$

$$Z'(0) = 1 + \frac{\eta f}{\ell}\sum_i \beta_i \tau_i$$

The real part of Eq. (8.17) is given by

$$\text{Real}\,[n_1(t)] = \frac{n_o\delta}{\ell}\,\frac{\cos(\omega t - \phi)}{\left[\left(\frac{\eta f}{\ell}\sum_i \frac{\omega^2\tau_i^2\beta_i}{\omega^2\tau_i^2+1}\right)^2 + \left(\omega + \frac{\eta f}{\ell}\sum_i \frac{\omega\beta_i\tau_i}{\omega^2\tau_i^2+1}\right)^2\right]^{1/2}} \qquad \text{Eq. (8.18)}$$

$$\tan\phi = \frac{\omega + \frac{\eta f}{\ell}\sum_i \frac{\omega\beta_i\tau_i}{\omega^2\tau_i^2+1}}{\frac{\eta f}{\ell}\sum_i \frac{\omega^2\tau_i^2\beta_i}{\omega^2\tau_i^2+1}}$$

Consider two cases:

Case I: $\omega\tau_i \ll 1$

In this case Eq. (8.18) reduces to

$$\text{Real}\,[n_1(t)] = \frac{n_o\delta}{\left[\ell + \eta f \sum_i \beta_i\tau_i\right]}\,\frac{\sin\omega t}{\omega} \qquad \text{Eq. (8.19)}$$

Case II: $\omega\tau_i \gg 1$

In this case Eq. (8.18) reduces to

$$\text{Real}\,[n_1(t)] = \frac{n_o\delta\cos(\omega t - d)}{\ell\left[\omega^2 + \left(\frac{\eta f\beta}{\ell}\right)^2\right]^{1/2}} \qquad \text{Eq. (8.20)}$$

where

$$\tan d = \frac{\omega\ell}{\eta f\beta}$$

Thus, for extremely low or high frequencies, the neutron response behaves as $1/\omega$.

In conducting this experiment it sometimes turns out to be simpler experimentally to have the reactivity vary as a step function rather than as a sine function. This complicates the analysis somewhat. First the reactivity variation is analyzed into a Fourier series.

$$\rho(t) = \frac{4\delta}{\pi}\sum_{\substack{j=1\\ j\ \text{odd}}}^{\infty}\frac{\sin j\omega t}{j} \qquad \text{Eq. (8.21)}$$

This is substituted in Eqs. (8.14) and (8.15) which are solved as before. The result is

$$n_1(t) = \frac{4n_o\delta}{\pi\ell}\sum_{\substack{j=1\\ j\ \text{odd}}}^{\infty}\frac{\sin(j\omega t - \phi_j)}{j}\,\frac{1}{\left[\left(\frac{\eta f}{\ell}\sum_i \frac{(j\omega)^2\tau_i\beta_i}{(j\omega)^2\tau_i^2+1}\right)^2 + \left(j\omega + \frac{\eta f}{\ell}\sum_i \frac{j\omega\beta_i\tau_i}{(j\omega)^2\tau_i^2+1}\right)^2\right]^{1/2}}$$

Eq. (8.22)

The particular result obtained from Eq. (8.22) with a cycle time of 5 seconds, was used for calibration purposes. For this period

$$n_1(t) = \frac{4n_o\delta}{\pi\ell} \sum_{\substack{j=1 \\ j\ \text{odd}}}^{\infty} \frac{1}{j} \frac{\sin(j\omega t - \phi_j)}{\left[\left(\frac{\eta f \beta}{\ell}\right)^2 + \left(j\omega + \frac{\eta f}{\ell} \sum_i \frac{\beta_i}{j\omega\tau_i}\right)^2\right]^{1/2}} \qquad \text{Eq. (8.23)}$$

where

$$\tan \phi_j = \frac{j\omega + \frac{\eta f}{\ell} \sum_i \frac{\beta_i}{j\omega\tau_i}}{\frac{\eta f \beta}{\ell}}$$

The initial rod oscillating experiments were done with a simple square-wave rod oscillator consisting of an air cylinder hung from the control rod drive, and with the control rod connected to the piston shaft. Air was applied to the cylinder to lift the rod. The rod returned by gravity. Limit stops on the piston shaft allowed the amplitude of oscillation to be selected. A linear differential transformer provided a signal, which was recorded, to indicate the oscillating component of the rod position. The average rod position was shown by the selsyns, which were part of the rod drive unit.

The air to the oscillating unit was controlled by a quick-acting, three-way solenoid valve located within a short distance of the drive unit, in order to minimize air travel time. The valve was operated by a cam switch driven by a timing motor. Transit time of the rod was very short (approximately 0.05 second) compared with the time at rest, so that the net motion was essentially a square wave.

Boron-lined, gamma-compensated ionization chambers were used exclusively for data taking. For the principal source of data, one of these chambers fed into a Beckman logarithmic amplifier and then into a Sanborn recorder. With this recorder, it was possible to suppress the zero and use the recorder at maximum sensitivity to observe the oscillating component of the power level. Since the logarithmic amplifier gave the fractional change in power level, it was not necessary to record the amount of zero suppression in the recorder.

The experimental procedure was as follows: The rod oscillator was preset for the desired amplitudes. Two amplitudes were used, --1/2 inch and 1-1/2 inches. The oscillator frequency was adjustable at the control console. The reactor was made critical when the control rod was in the appropriate position. The rod was lowered a distance equal to one-half the oscillating amplitude, and the oscillator was turned on. Usually 10 to 20 oscillations allowed a good average value to be obtained for the peak-to-peak amplitude of the power level. The oscillator was turned off when the rod was in the down position.

Figure 8.3 gives the variation of the signal amplitude with the frequency of the rod oscillator. The cycle times for this curve range from 2.5 seconds to 180 seconds. The oscillation amplitude of the rod was maintained constant throughout the range. The low frequency end of the curve shows the tendency toward a $1/\omega$ response as the analysis suggests. In the studies that employed this method of calibrating control rods, a time cycle of 5 seconds was used.

In applying the technique, a relative or absolute method may be used. In the relative method the reactivity calibration is obtained by taking one or more rod-bump period measurements. From this, the constant relating the oscillating component of reactivity to the detector

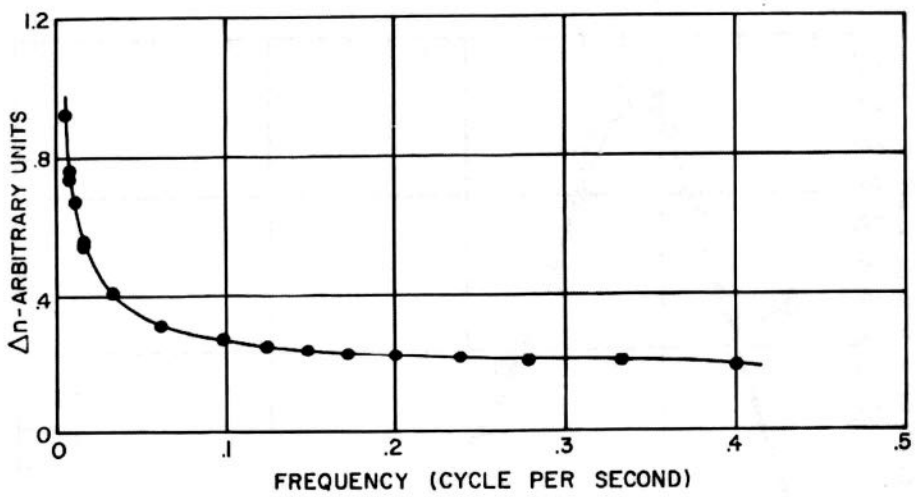


Fig. 8.3 - Frequency Response of Neutron Level to Reactivity Oscillations.

amplitude response is evaluated. In the absolute method, it is necessary to evaluate Eq. (8.23). This was done by choosing appropriate values for the parameters and summing the terms between 1 and 49. For the particular evaluation the parameters used were

$$\beta = 0.73$$
$$\eta = 2.09$$
$$\ell = 10^{-4} \text{ seconds}$$
$$f = 0.58$$

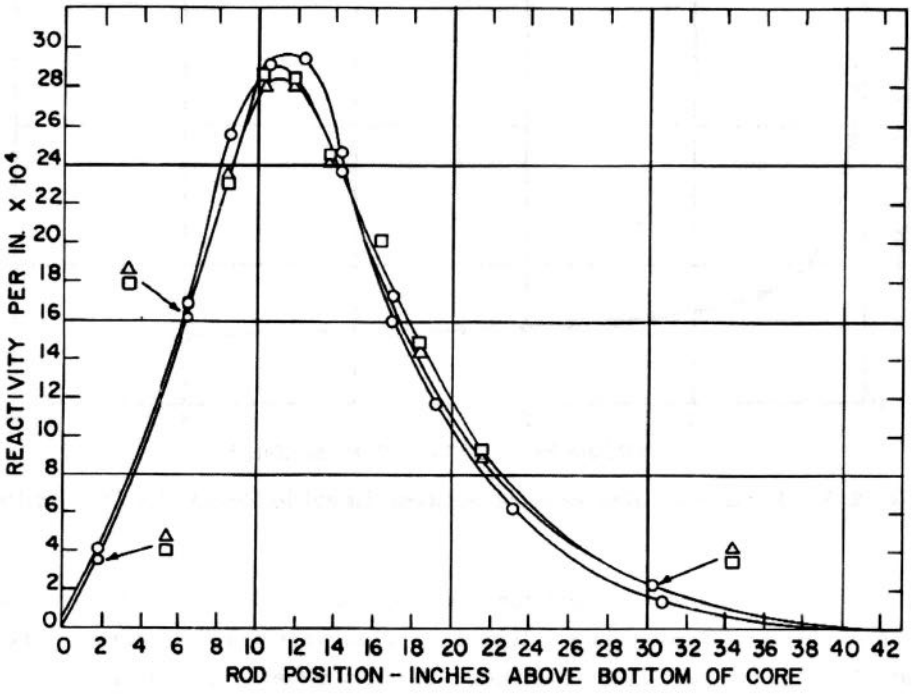
The value chosen for ℓ has only slight effect on the evaluation. A change in ℓ by a factor of two changes the result by only 3 per cent.

The absolute calibration is quite sensitive, however, to the value chosen for f. The value of f was determined by a simple criticality calculation in which the homogeneous thermal poison required for criticality was determined. Using this poison cross section, a value was calculated for f. The uncertainty in f is estimated to be about 5 per cent.

The recorder chart, giving the amplitude of the oscillating component of the flux, introduces an uncertainty of about 2 per cent. In the relative method of calibration the period-bump technique is estimated to have an error or 2 per cent. This, combined with the chart error, gives an estimated 3 per cent error in the relative calibration technique. This is substantially smaller than the estimated error in the absolute calibration using rod oscillation, since the 5 per cent error in f, when combined with the chart error, leads to an estimate of 6 per cent total error.

Calibration curves obtained by using the above procedure and analysis are shown in Fig. 8.4. These were derived by use of the period method and the rod oscillator method, both relative and absolute. All three curves fall within the assigned uncertainties. The integrated rod worth values are in considerably better agreement than might be expected.

The rod oscillation technique of control rod calibrations has a number of important advantages. Perhaps the most significant is that it is about four times faster than the period rod-bump technique. In addition, less data reduction is required and fewer and less stable electronic units are required.



Legend	Integrated Worth
○ Rod-Bump Period	408×10^{-4}
△ Rod Oscillator (Relative)	402×10^{-4}
□ Rod Oscillator (Absolute)	408×10^{-4}

Fig. 8.4 - Calibration of a Cd-Ag Xenon Control Rod.

(e) Rod Drop Method. The rod drop method of control rod calibration has been used extensively. In certain applications difficulties have been encountered. Primarily these are associated with attempts to measure large reactivity changes. In practice it has been found that the rod drop and rod-bump period methods give comparable results if the reactivity change measured is relatively small, i. e., a fraction of β. If large, the change in flux distribution associated with the large reactivity change contradicts the assumptions which are implicit in the theory of the method.

A simple theory of the rod drop experiment may be derived from the reactor kinetic equations.[3] If a reactivity change ρ is introduced into a reactor suddenly ($\rho < \beta$), the flux level changes rapidly from its initial value ϕ_o to a new value ϕ. This change takes place before the delayed neutron emitters come into equilibrium at the new flux level. The flux ratio associated with the prompt drop (or jump) is

$$\frac{\phi}{\phi_o} = \frac{\beta(1 - \rho)}{\beta - \rho} \qquad \text{Eq. (8.24)}$$

If the relationship

$$\rho = \frac{k_{eff} - 1}{k_{eff}}$$

is introduced, and if the reactor is just critical when the flux is $\phi_o (k_{eff} = 1)$, the following relationship holds

$$\frac{\phi - \phi_o}{\phi} = \frac{\Delta k}{\beta}(1 - \beta) \qquad \text{Eq. (8.25)}$$

where $\Delta k = (k_{eff} - 1)$ is the change in k_{eff} introduced by the rod drop.

In applying this teqhnique a chart record is made of the neutron detector level as a function of time, starting immediately preceding the drop. The level ϕ_o is proportional to the reading immediately prior to the rod drop, while ϕ is proportional to the reading immediately afterward. The chart record following the drop may be extrapolated back to the time of the drop to obtain an accurate value corresponding to ϕ. Alternatively, the reading at a fixed time following the drop may be taken and the level corresponding to ϕ calculated by use of the known delayed neutron yields and periods.

8.18 TEMPERATURE COEFFICIENT OF REACTIVITY

The temperature coefficient of reactivity is an important reactor characteristic. Despite several schemes, good measurements of temperature coefficients are difficult to make. Four methods of measurement are outlined below. The best method for a given experiment will depend on the reactor characteristics and the type of instrumentation available.

(a) Constant-Temperature Calibrated-Rod Method. In applying this method, the reactor is made critical at some given temperature, with a single calibrated rod used to adjust the reactor. The position of the rod is noted for several temperatures with the reactor critical. The temperature coefficient is derived from the known reactivity change and the measured temperature change. The value derived for the temperature coefficient is usually assigned to the mid-point of the temperature range from which it was derived.

The principal difficulty with this type of measurement is achieving adequate stability of temperature. This may require rather elaborate systems of heating and mixing of the water.

This measurement requires that the rod being used to compensate for temperature change be calibrated. With substantial temperature coefficients of reactivity, or fairly large temperature ranges, the rod calibration may vary with temperature. To avoid difficulties, the rod may be calibrated as the measurement proceeds. The convenient way to do this is by the rod oscillator technique. If this is not available, the rod-bump period technique may be used, but this requires special techniques in order to compensate for temperature drifts during the measurement.

(b) Constant-Temperature Period Method. This is a modification of procedure (a), and is particularly useful in cases where the magnitude of the temperature coefficient is small. With the reactor at a constant uniform temperature, the reactivity is made positive and determined by measuring the reactor period and calculating the reactivity from the inhour formula. After the reactor period is measured, the power is reduced by the motion of one or more rods which can be repositioned accurately. The reactor is next heated to a new temperature; when this becomes constant and uniform, the control rods are repositioned exactly to their previous position. The reactor period, which is still positive, is again measured and the corresponding reactivity calculated. The difference in the two reactivity values, divided by the change in temperature, will give the value of the temperature coefficient.

(c) Temperature-Drift Power-Drift Method. In applying the power drift technique to measuring temperature coefficients, the temperature of the reactor is allowed to change slowly, and both temperature and reactor power are measured continuously. In principle, the temperature may be either rising or falling. In the experiments attempted, it has been found expedient to heat the reactor to an elevated temperature and then cool it by allowing heat loss to the room or by circulating cold water through the steam coils in the heat exchanger. These procedures have resulted in a nearly uniform drop of a few degrees centigrade per hour, --a change that can be measured accurately and that permits slow changes in power, which may also be measured accurately. During the experiment, the control rods are held fixed.

With a negative temperature coefficient, as the temperature falls, the reactivity slowly increases, and the reactor power rises on an increasing exponential curve. Analysis of the power curve obtained yields the rate at which the reactivity increases. This quantity, coupled with the rate at which the temperature decreases, yields the temperature coefficient for a particular temperature and rod configuration. The temperature coefficient is given by the equation

$$k(T) = \frac{\partial \rho}{\partial t} \Big/ \frac{\partial T}{\partial t} \qquad \text{Eq. (8.26)}$$

where $k(T)$ = temperature coefficient of reactivity
T is the temperature
ρ is the reactivity
t is the time.

The rate at which the temperature changes with time is measured directly. The rate at which reactivity changes with time may be expressed in terms of reactor power or neutron level. One method by which this can be done is to deduce the reactor period as a function of time. On the assumption that this is the asymptotic period, the reactivity as a function of time may be calculated from the inhour equation. The two functions of time, reactivity and temperature, yield the coefficient.

(d) Temperature-Drift Constant-Power Method. A modification of the above procedure, which depends more on experiment and less on theoretical analysis of the data, consists of the following procedure: The temperature of the reactor is allowed to change slowly as above, but the reactor power is held constant by continued adjustment of a calibrated control rod. The reactor temperature and the position of the control rod are noted at frequent intervals. The position of the rod as a function of time also allows an experimental determination of the rate of change of reactivity as a function of time. This is divided by time rate of change of temperature at any given time to yield a value of the temperature coefficient.

THE MEASUREMENT OF EXCESS REACTIVITY AND SHUTDOWN REACTIVITY

by R. T. Bayard and W. H. Hartley

8.19 INTRODUCTION

Two of the more important measurements made on a mock-up critical assembly are those which (1) allow an inference of the excess reactivity of the reactor with all control rods withdrawn, and (2) those which measure the degree of subcriticality of the reactor in its shutdown condition with all control rods inserted.

The excess reactivity measurement is used in conjunction with a design calculation to determine whether the reactor design is satisfactory. There are, in effect, two steps in the application of this measurement to the design procedure. In the first step, the cold mock-up results are used to check the degree to which the design procedure has succeeded in predicting the critical configuration. This amounts to establishing the amount of uncertainty in the design procedure. The second step is to extend the design procedure to the hot, end-of-life condition in order to determine whether the design is satisfactory.

The shutdown reactivity can be estimated by extrapolation of rod bank calibrations, as previously described. Another method, which is particularly applicable to tall thin cores such as the PWR, involves a measurement of the buckling in the subcritical rodded portion of a critical core. From this the k_{eff} of the core with rods fully inserted may be deduced.

8.20 DIFFERENTIAL WATER HEIGHT EXPERIMENT

The differential water height experiment is one in which the reactor is made critical with the water level at less than the full core height. The measurement consists essentially of two parts: First, the measurement of the critical height of the water; second, the reactivity worth of the water per unit change in water height. The theory of the method depends upon the separability of the axial leakage of neutrons from the transverse leakage. For tall thin reactors, such as the PWR seed, axial and radial separability would appear to be quite reasonable; and it, therefore, seemed quite promising to apply this technique to the PWR seed and blanket reactor.

As already noted, the basic assumption made in deriving the theory of this experiment is that the neutron leakage is separable into an axial component and a transverse component. The separability may be expressed analytically by writing

$$B^2 = B_r^2 + B_z^2 \qquad \text{Eq. (8.27)}$$

where B^2 is the total buckling of the reactor, B_r^2 is the transverse buckling, and B_z^2, the axial buckling, is given by

$$B_z^2 = \left(\frac{\pi}{h + \lambda_z}\right)^2 \qquad \text{Eq. (8.28)}$$

where h is the reactor height and λ_z the total axial reflector savings. If a one-group expression is written for the reactivity,

$$\rho = \frac{k_{eff} - 1}{k_{eff}} = \frac{k_\infty - (1 + M^2B^2)}{k_\infty} \qquad \text{Eq. (8.29)}$$

where M^2 is the migration area; and if this expression is differentiated to form $\frac{\partial \rho}{\partial z}$, the result is

$$\frac{\partial \rho}{\partial z} = - \frac{M^2}{k_\infty} \frac{\partial B^2}{\partial z} = \frac{2\pi^2 M^2}{H^3 k_\infty} \qquad \text{Eq. (8.30)}$$

where H, the equivalent bare reactor height, is equal to $(h + \lambda_z)$. Since $\frac{M^2}{k_\infty}$ is a property of the reactor material and is independent of reactor height, a plot of H versus $\left(\frac{\partial \rho}{\partial z}\right)^{-1/3}$ should be a straight line according to this model. It is a necessary condition for the applicability of this model that the experimental data should fit a straight line. Two sets of data will be presented to illustrate the results on the PWR Flexible Assembly.

Figure 8.5 shows data taken with a slab assembly. The reactor consisted of a slab seed 6-inches thick, 36 or more inches long, and 6 feet in total height. PWR blanket subassemblies were located adjacent to the sides of the seed. The blanket thickness was either 6 or 12 inches, depending upon the number of rows of blanket material used. Control rods were available for use in each of the seed clusters. The slab was made critical at different heights by several methods. One method was by the variation of the number of rows of blanket material placed next to the seed; another was by the variation of the length of the slab; and a third involved complete insertion of selected control rods in the slab. It was found experimentally that the moderator worth $(\partial\rho/\partial z)$ versus height curves obtained by these methods were equivalent.

The solid line plot in Fig. 8.5 is a least squares fit to the experimental points. No systematic deviation from the straight line can be seen; and so it must be assumed that the model is reasonable. *

The same type of experiment was performed for the complete PWR mock-up. In this case the reactor was made critical at various heights by inserting selected control rods into the reactor core. The data are shown on Fig. 8.6. Again the solid line is a least squares fit to the experimental data.

Figures 8.5 and 8.6 can be considered to represent verification of the model insofar as its applicability to the particular configurations tested is concerned. As was noted previously, the data for Fig. 8.5 were obtained by varying the amount of blanket material, the length of the slab, or the control rod pattern. Points taken by the different methods are essentially indistinguishable from each other. Physically, the quantity varied by each of these methods of varying reactivity is the buckling in the transverse direction. The changes in the transverse

* It should be noted, however, that if the reflector savings is a linear function of height (i.e., $\lambda_z = a + bz$), the data will still fit a straight line, but the slope will not be $\left(\frac{2\pi^2 M^2}{k_\infty}\right)^{1/3}$.

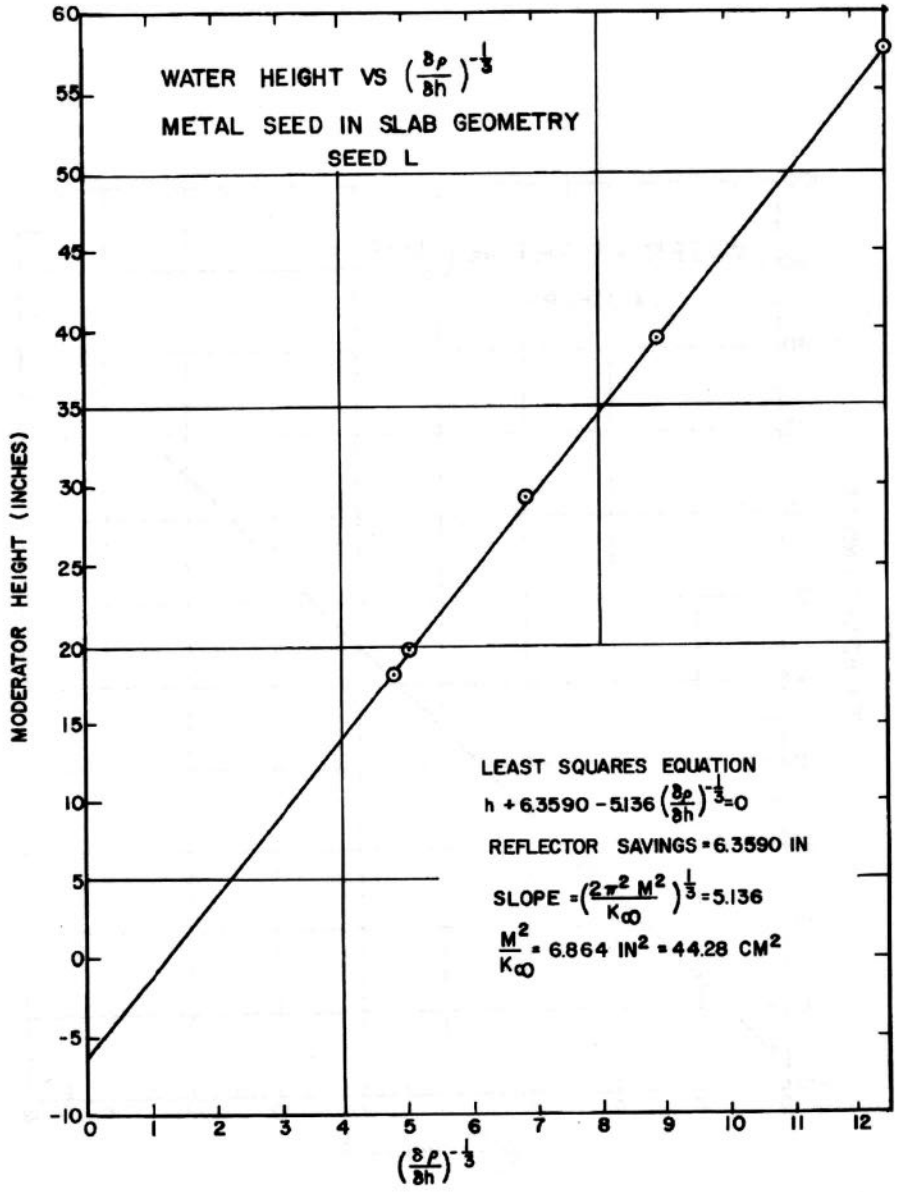


Fig. 8.5 - Plot of Differential Water Height Data, Slab Reactor

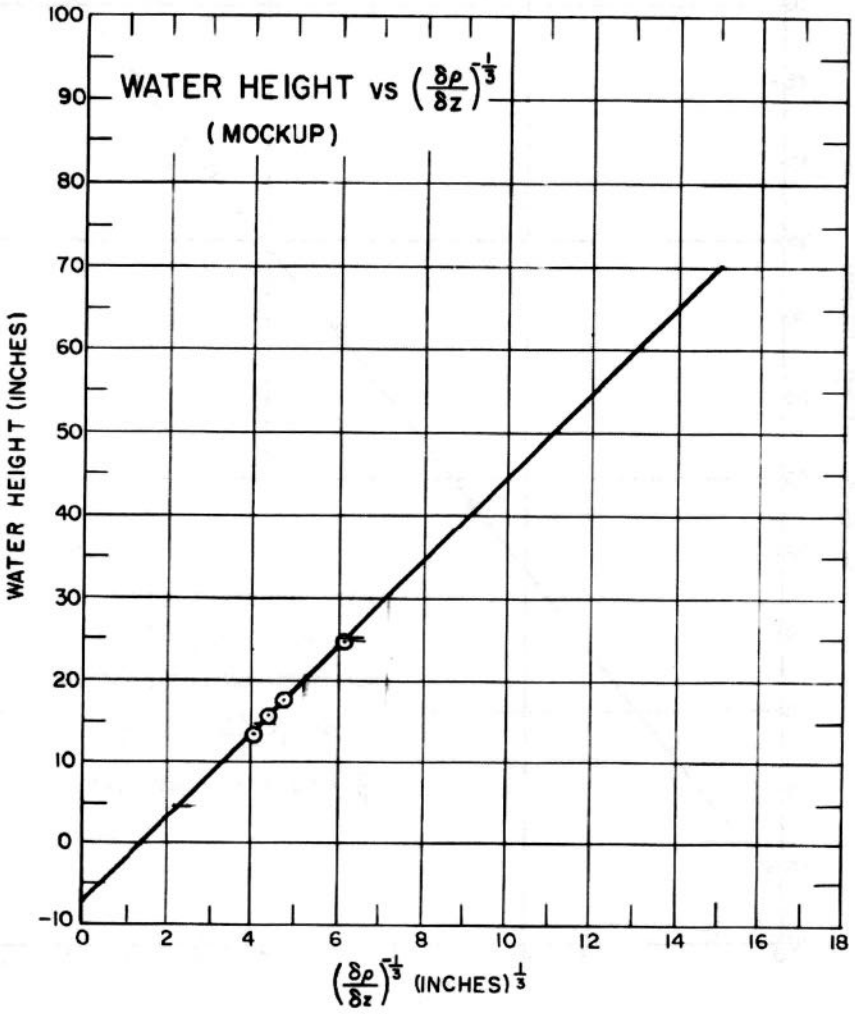


Fig. 8.6 - Plot of Differential Water Height Data; Mock-up Reactor

buckling between different core configurations are compensated by a vertical buckling change to maintain criticality. The experimental data verify a specific prediction of the model, namely, that the reactivity worth of the water increment is a function only of height. The verification refers specifically to a reactor in which criticality is maintained at various moderator heights by appropriate adjustment of the transverse buckling.

If, instead of adjusting the buckling to maintain criticality, we now imagine the reactor to be made substantially supercritical by raising the moderator height at constant transverse buckling, the resulting excess reactivity can be obtained by integration of the differential curve. In particular, the excess reactivity of the reactor with all rods removed is determined by integration of the differential moderator worth curve from criticality to full reactor height.

In Eq. (8.30) the value of $\frac{M^2}{k_\infty}$ is a property of the reactor material. For relatively uniform reactors the value of this quantity could be calculated and the slope of the H vs $\left(\frac{\partial \rho}{\partial z}\right)^{-1/3}$ line determined. A single experimental measurement which establishes a point on the curve, together with the slope, completely determines the curve. Alternatively, a series of experimental points may be considered a measurement of $\left(\frac{M^2}{k_\infty}\right)$.

For reactors such as the PWR seed and blanket design, which are of non-uniform composition in the transverse plane, $\frac{M^2}{k_\infty}$ cannot be calculated. Measurements such as those given in Figs. 8.5 and 8.6 not only verify the simple model being used to interpret the experiment, but also provide an effective value of $\left(\frac{M^2}{k_\infty}\right)$. From the data taken to construct these curves, curves of differential moderator worth may be obtained. These are given in Figs. 8.7 and 8.8. Figure 8.7 is derived from Fig. 8.5, and Fig. 8.8 from Fig. 8.6.

One final point concerning the water height experiment should be noted. In performing the experiment, the upper axial reflector consists of unmoderated fuel-bearing material at all partial heights. To minimize the multiplication of this zone, the control rods are positioned just above the water surface at all partial heights. The separation between control rod tips and the water is carefully adjusted at each partial height. Thus the effectiveness of the fueled reflector is reduced and is, in any event, maintained constant at all but full water height. At full height the effect of a small change in reflector savings on the total buckling is insignificant in reactors such as the PWR.

8.21 SUBCRITICALITY MEASUREMENTS BY NEGATIVE BUCKLING

The technique of subcriticality measurements through negative buckling is, like the differential water worth measurements, one which depends on the separability of the axial and transverse components of the buckling. Consider a reactor which is made critical by uniform withdrawal of the control rods as a bank. In the region below the control rods the reactor leaks neutrons transversely into the reflector and axially into the reflector at the bottom of the reactor, and also into the rodded region in the upper portion of the reactor. If the upper portion of the reactor is reasonably homogeneous with respect to distribution of control rods, the axial distribution of flux will be a negative exponential over most of the rodded region. The upper region will act like an exponential experiment, with the principal reacting zone of the reactor below the rods providing the neutron source (this assumes that the rodded portion

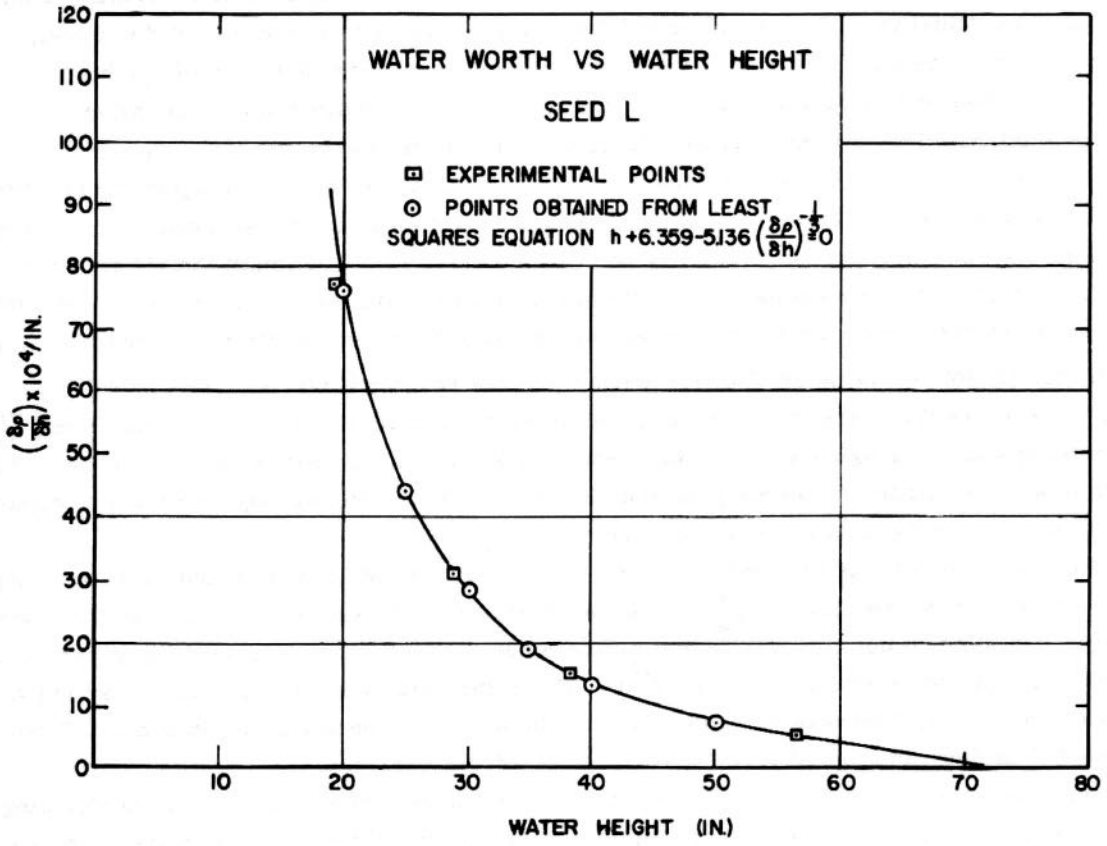


Fig. 8.7 - Differential Water Worth versus Height for Slab Reactor.

of the reactor has an effective multiplication constant of less than one with the control rods present).

An axial flux distribution taken through the reactor will have a shape which is generally a cosine in the reacting portion of the reactor and which is a negative exponential in the upper portion of the reactor. Figure 8.9 is such a flux distribution. The portion of the flux map in the rodded region can be fitted by a least squares technique to a function of the form.

$$\phi = \phi_o \exp(-B_z Z)$$

by the use of a special IBM-704 code, the EXFIT code (see Appendix F). From this fit, the axial buckling B_z is obtained.

The procedure by which the shutdown k_{eff} is determined from the B_z can be explained by considering its application to a slab reactor. As a first step, a set of few-group diffusion constants are computed for the seed and blanket region of the core. These are obtained through the MUFT code (see Appendix E). In the rodded region of the core the rod material may be treated as additional scattering metal (zirconium) or neglected, i.e., treated as void. The results are not sensitive to this choice, and for simplicity, we have preferred the latter. The neutron absorption properties of the control rods are accounted for by the addition of distributed poison in the seed region. This poison may be taken as a pure thermal poison, or

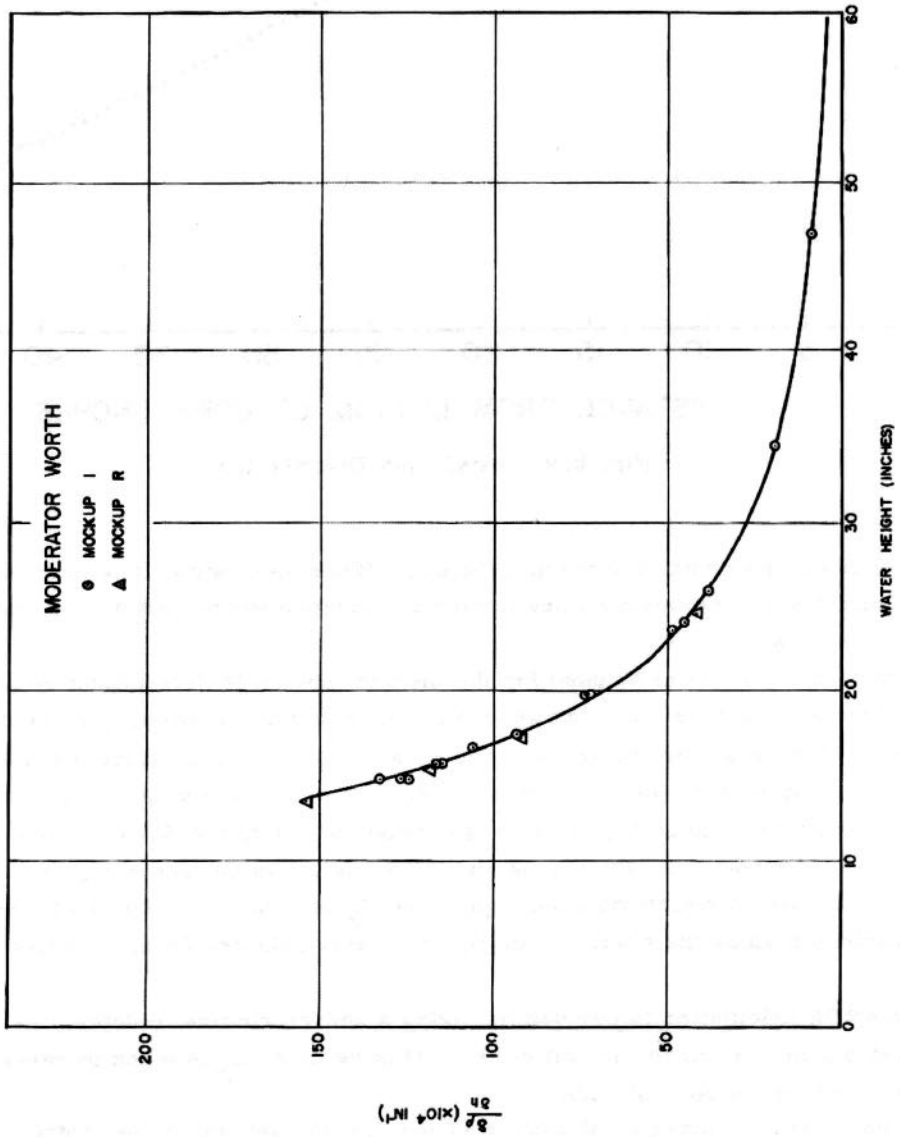


Fig. 8.8 - Differential Water Worth versus Height for PWR Mock-up.

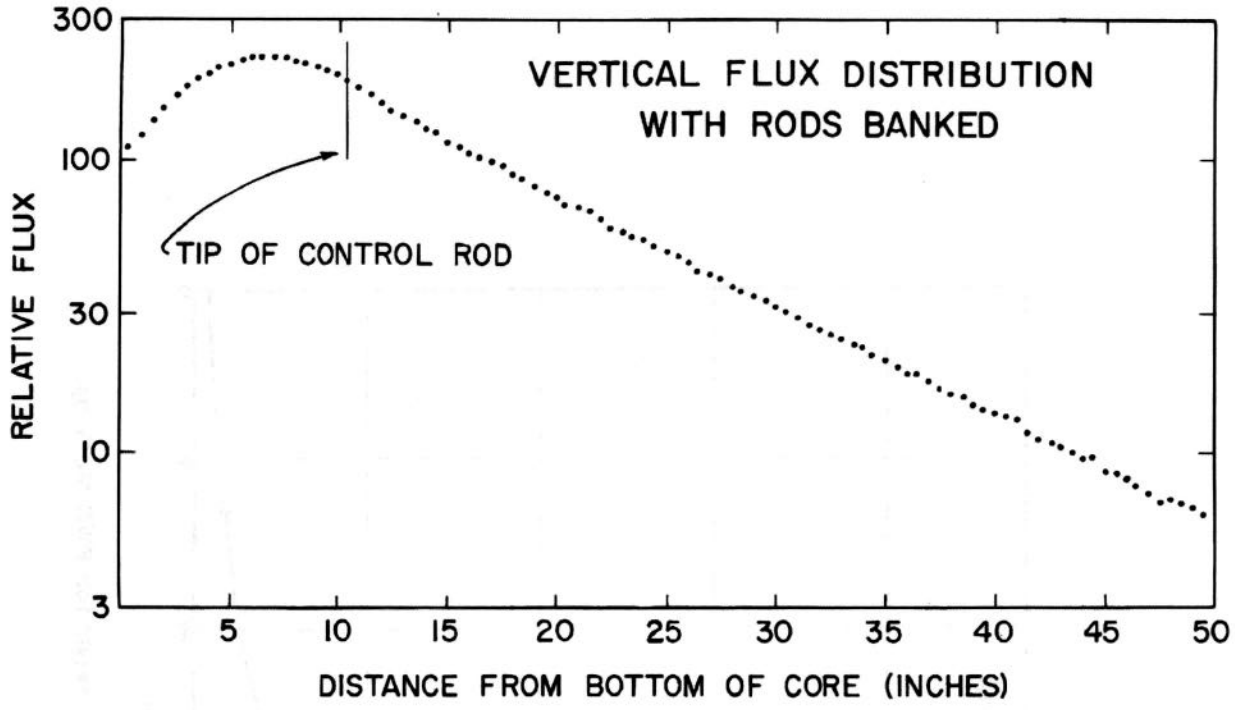


Fig. 8.9 - Axial Flux Distribution.

separated into an epithermal and thermal poison. When separated, it is necessary to use either an empirically determined value for the fast absorbtion or to set it as some fraction of the thermal group absorbtion.

The appropriate value to be used for the thermal poison is determined by calculation, using the thermal poison as the unknown in a one-dimensional criticality calculation. A slab calculation is carried out by taking the x coordinate, for example, along the thin dimension of the seed, the y coordinate along the length of the slab, and the z axis as the vertical dimension. A criticality calculation in slab geometry is carried out using the WANDA code (see Appendix A). The calculation is made along the x-axis, with the spatial variations along the y and z axes accounted for by use of measured bucklings. The B_y is determined by measurement of the cosine distribution along the y axis. For B_z, the value obtained from the exponential fit is used.

The WANDA calculation is carried out using a search routine to determine Σ_p, the thermal group added poison, required for criticality. This value of Σ_p is taken to represent the thermal absorption worth of the control rods.

The final step is to make a calculation of the reactor assuming the control rod bank is fully inserted and that the rods are represented by the same Σ_p. The axial buckling is taken to be positive and that corresponding to the full core height

$$B_z^2 = \left(\frac{\pi}{H}\right)^2$$

where H is the full core height plus total axial reflector savings. The reason for this choice

is that with the rods fully inserted the axial shape becomes a full cosine, and the net axial leakage from the rodded core is now outward. A WANDA calculation similar to the first is performed along the x-axis with this new axial buckling (and the same B_y^2 as before) and the eigenvalue determined. This eigenvalue is the effective multiplication constant of the reactor with all rods fully inserted.

It has been found in practice that it is not necessary to find the exact Σ_p corresponding to the control rod worth which makes the eigenvalue of the first problem unity. The ratio of the eigenvalues of the two problems is essentially constant for a rather wide range of reactor constants including Σ_p. The ratio of the second problem eigenvalue to the first problem eigenvalue is the desired shutdown k_{eff}.

This method has been found effective in analyzing a variety of seed and blanket critical assemblies on the PWR.

INHOUR EQUATION CORRECTIONS

by W. H. Hartley

8.22 INTRODUCTION

In many reactor experimental techniques the primary experimental datum is a reactor period. The period is converted to a reactivity value through use of the well-known inhour equation.[9] This procedure can be the source of significant errors.

The experimental technique which is particularly vulnerable in this respect is the differential water-height measurement. In this experiment, the primary datum is the incremental reactivity per unit change in water height. If the conversion of period to reactivity is in error, the interpretation of the experimental data will also be in error. Other measurements which are affected by this uncertainty are rod calibrations (and any deductions made from these calibrations) and temperature coefficient measurements.

There are two distinct and different problems with the use of the inhour equation. The first is associated with the uncertainties in the data on the total yield of delayed neutrons in fission and the distribution of this yield among the various delayed groups. The second is the solution of the reactor kinetic equation, taking into account the energies of the delayed neutrons and the possibility of fast fission in the reactor.

8.23 DELAYED NEUTRON DATA

For many years, the accepted data on delayed neutron yields were those of Hughes and his collaborators.[4] More recently, Keepin at Los Alamos has reported results which are distinctly different from those of Hughes.[5] The total yield of delayed neutrons per neutron emitted in fission, β, is reduced from 0.00755, quoted by Hughes, for uranium-235 to approximately 0.0070 reported by Keepin. Another significant change is in the distribution of the yield among the various groups.

Since the product of reactivity and period is approximately proportional to β for long periods, the reactivity value associated with a given period will change proportionally to the change in β. The actual change will depend in detail on the reactor period and the distribution of delayed neutrons in the various groups.

8.24 FORM OF THE INHOUR EQUATION

Generally, the inhour equation is derived for a bare reactor with a one-group treatment. If account is taken of the fact that the delayed neutrons are emitted at lower energies than the fission neutrons, it is clear that these neutrons will be more effective. This enhances the effective value of β.

If uranium-238 is present, several additional factors enter. One is that uranium-238 emits more delayed neutrons in fission than uranium-235. Another is the fast fission effect in uranium-238. These effects tend to compensate, and the precise effect will depend upon the distribution of the uranium-238 in the reactor.

It is possible, however, to derive an equation in the form of the usual inhour equation, which includes the effects of U^{238} fissioning and the difference in age between delayed and prompt neutrons. Such an equation has the form

$$\rho = \frac{\Lambda}{T} + \sum_i \frac{\bar{\beta}_i t_i}{T + t_i} \qquad \text{Eq. (8.31)}$$

where

Λ = prompt neutron lifetime

T = reactor period

t_i = mean life of i^{th} group of delayed neutrons

$\bar{\beta}_i$ = effective fraction of delayed neutrons in the i^{th} group.

It may be shown[6, 7, 8] that $\bar{\beta}_i$ is the change in reactivity produced by the insertion of the i^{th} group of delayed neutrons, properly distributed spatially into a critical reactor. It is thus possible to calculate $\bar{\beta}$ by the following technique:

With the use of the WANDA code (see Appendix A), a four-group, one-dimensional criticality calculation is made in the usual manner. The results of this calculation give the unperturbed eigenvalue (λ_t) and the spatial distribution of delayed neutron precursors from uranium-235 fission. The distribution of delayed neutron precursors from uranium-238 fission may be obtained by observing that fast fission takes place only in the first group of neutrons. Thus the spatial distribution of neutrons in the i^{th} group may be computed. Using these delayed neutrons as the source, a second criticality type calculation is made with diffusion constants and absorption cross sections appropriate for neutrons with an energy corresponding to the mean effective delayed neutron energy (in general we have used 0.5 Mev for all the delayed groups). The calculation (one iteration through the energy groups) gives the value of the effective multiplication constant (λ_d) of these delayed neutrons in one generation. Assuming that the ratio λ_d/λ_t is constant regardless of the absolute values of the two eigenvalues, the change in reactivity produced by insertion of the delayed neutrons is

$$\rho = \frac{\lambda_d - \lambda_t}{\lambda_d} \qquad \text{Eq. (8.32)}$$

and this is $\bar{\beta}$. The values of $\bar{\beta}$ obtained in this manner for the PWR slab critical experiments and for the PWR mock-up experiment are approximately 16 per cent higher than the uncorrected β values.

In general, the experimental data reported in this volume which involve translation of period to reactivity have been carried out using Hughes's data and the uncorrected form of the

inhour equation. Where the corrections are important they have been made. The fact that corrections have been made is noted in connection with the quoted data.

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